Dated at Rockville, Maryland this 15th day of May 1996.

James Lieberman,

Director, Office of Enforcement.

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UNITED STATES NUCLEAR REGULATORY COMMISSION

Biweekly Notice

Applications and Amendments to Facility Operating Licenses Involving No Significant Hazards Considerations

I. Background

Pursuant to Public Law 97-415, the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. Public Law 97-415 revised section 189 of the Atomic Energy Act of 1954, as amended (the Act), to require the Commission to publish notice of any amendments issued, or proposed to be issued, under a new provision of section 189 of the Act. This provision grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from April 27, 1996, through May 10, 1996. The last biweekly notice was published on May 8, 1996 (61 FR 20842).

Notice Of Consideration Of Issuance Of Amendments To Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, And Opportunity For A Hearing

The Commission has made a proposed determination that the following amendment requests involve no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The basis for this proposed determination for each amendment request is shown below.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received before action is taken. Should the Commission take this action, it will publish in the Federal Register a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Chief, Rules Review and Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and should cite the publication date and page number of this Federal Register notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By June 21, 1996, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1-(800) 248-5100 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to (Project Director): petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this Federal Register notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for a hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room for the particular facility involved.

Baltimore Gas and Electric Company, Docket Nos. 50-317 and 50-318, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of amendments request: April 5, 1996

Description of amendments request: Pursuant to 10 CFR 50.80 and 50.90, the Baltimore Gas and Electric Company (BGE) hereby requests the transfer and amendment of Operating License Nos. DPR-53 and DPR-69 for Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2.

The proposed license transfers and amendments are requested as part of the pending merger between BGE and Potomac Electric Power Company into Constellation Energy Corporation. The proposed license transfers would transfer authority to possess and operate Calvert Cliffs from BGE to Constellation Energy Corporation. The proposed amendments would change the licenses as well as the related Technical Specifications, to reflect this transfer by submitting Constellation Energy Corporation in place of BGE as the licensee for Calvert Cliffs.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendment will change the name of the licensee authorized to possess and operate Calvert Cliffs Nuclear Power Plant from Baltimore Gas and Electric Company (BGE) to Constellation Energy Corporation. This amendment request is necessary because of a proposed merger of BGE and Potomac Electric Power Company into Constellation Energy Corporation. As a result of the savings achieved through a reduction in operating costs due to the merger, Constellation Energy Corporation will have the financial resources to possess and operate Calvert Cliffs.

In addition, Constellation Energy Corporation personnel will be technically qualified to operate the plant. Baltimore Gas and Electric Company nuclear personnel have been named to management positions in Constellation Energy Corporation, and will remain responsible for Calvert Cliffs operation and maintenance. The proposed amendment involves no changes in the training program or operating organization for Calvert Cliffs.

The proposed amendment does not require any physical change to the facilities or substantive modifications to the Technical Specifications or to procedures. The proposed change does not increase the probability of an accident previously evaluated because it does not affect any initiators in any previously evaluated accidents. The proposed change does not increase the consequences of an accident previously evaluated because it does not affect any of the items on which the consequences depend.

Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendment does not modify the plant's configuration or operations. As a result, no new accident initiators are introduced. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Would not involve a significant reduction in a margin of safety.

This amendment request is necessary because of a proposed merger of BGE and Potomac Electric Power Company into Constellation Energy Corporation. As a result of the savings achieved through a reduction in operating costs due to the merger, Constellation Energy Corporation will have the financial resources to possess and operate Calvert Cliffs. Also, Constellation Energy Corporation personnel will be technically qualified to operate the plant. Baltimore Gas and Electric Company nuclear personnel have been named to management positions in Constellation Energy Corporation, and will remain responsible for Calvert Cliffs' operation and maintenance. The proposed amendment involves no changes in the training program or operating organization for Calvert Cliffs. In addition, the proposed amendment to substitute Constellation Energy Corporation for BGE does not result in any changes to the physical design or operation of the plant. Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendments request involves no significant hazards consideration.

Local Public Document Room location: Calvert County Library, Prince Frederick, Maryland 20678.

Attorney for licensee: Jay E. Silbert, Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: Susan Frant Shankman, Acting

Carolina Power & Light Company, et al., Docket Nos. 50-325 and 50-324, Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina

Date of amendments request: April 2, 1996

Description of amendments request: The proposed amendments revise the Brunswick Steam Electric Plant, Units 1 and 2, Technical Specifications (TS) to allow uprate of the units to 105 percent of rated thermal power.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

. May the proposed activity involve a significant increase in the probability or consequences of an accident evaluated previously in the Safety Analysis Report?

The increase in power level, steam flow, feedwater flow and associated instrument setpoint changes will not significantly increase the probability or consequences of an accident previously evaluated.

The probability (frequency of occurrence) of Design Basis Accidents occurring is not affected by the increase in power level, as plant equipment will remain in compliance with the applicable regulatory criteria (ASME Codes, IEEE Standards, NEMA Standards, Regulatory Guide criteria, etc.). The physical plant changes necessary to support power uprate include instrument setpoint changes, indicating meter scale changes for the RWCU [reactor water cleanup] System flow and Main Steam Flow indicators, Leak Detection, Process Computer, ERFIS [emergency response facility information system], and Feedwater System software changes, and SRV [safety/relief valve] setpoint changes. The setpoints were calculated in accordance with the CP&L Setpoint Methodology. Utilizing this methodology ensures scram setpoints (instrument settings that initiate automatic plant shutdowns) will be established such that there is no significant increase in scram frequency due to uprate. No new challenges to safety related equipment will result from power uprate.

The changes in consequences of hypothetical accidents which would occur from 102% of the uprated power (2609 MWt), compared to those previously evaluated from [greater than or equal to] 102% of the original power (2485 MWt), are not significant, because the accident evaluations at uprated power will not result in exceeding the NRC approved acceptance limits. The spectrum of hypothetical accidents and transients has been investigated, and those accidents/transients currently evaluated in the UFSAR [Updated Final Safety Analysis Report] were shown to meet the plant's current regulatory criteria at uprated conditions (105%). In the area of core design, for example, the fuel operating limits will still be met at the uprated power level, and fuel reload analyses show plant transients will still meet the criteria accepted by the NRC as specified in NEDO-24011, ''GESTAR II.'' Challenges to fuel or ECCS [emergency core cooling system] performance have been evaluated and shown to meet the criteria of 10CFR50 Appendix K. Challenges to the containment have been evaluated and still meet 10CFR50 Appendix A Criterion 38, Long Term Cooling, and Criterion 50, Containment. Bounding events involving radiological releases have been evaluated and were shown to be well within the criteria of 10CFR100.

2. May the proposed activity create the possibility of a new or different kind of accident from any accident previously evaluated in the Safety Analysis Report?

The change in reactor thermal power will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Equipment that could be affected by power uprate has been evaluated. No new operating mode, safety related equipment lineup, accident scenario, or equipment failure mode was identified. The full spectrum of accident considerations defined in the BNP [Brunswick Nuclear Plant] UFSAR has been evaluated and no new or different kind of accident has been identified. Uprate uses developed technology and applies it within the capabilities of existing plant equipment in accordance with existing regulatory criteria including NRC approved codes, standards, and methods. General Electric has designed BWRs [Boiling Water Reactors] of higher power levels than the uprated power of any of the currently uprated BWR/4 fleet and has not identified new power dependent accidents.

The changes to the Technical Specifications required to implement power uprate make little change to the plant's configuration. These changes fall into three major categories. The first includes those changes resulting from power uprate parameter changes. These parameter changes, such as the increase in vessel pressure, temperature and piping system flows are minor in nature. The evaluations have shown the plant is still within its design capabilities when operating under these conditions. The changes required as a result of power uprate will not affect the design function(s) of currently installed equipment; therefore, there is no possibility of a new or different kind of failure mode. The second set of changes is a result of applying setpoint methodology to calculate TS Allowable Values and Normal Trip Setpoints for instruments that are directly affected by the parameter changes due to power uprate. By using CP&L's methodology, the TS values were calculated to ensure adequate margin exists between the analytical limit and the TS Allowable Value. The third change include [sic] setpoints that were reconstituted by the power uprate project. Again, CP&L methodology was applied and the results show the setpoints have moved to a more conservative value. This will reduce the likelihood of spurious scrams and unnecessary challenges to safety systems while ensuring initiation/actuation equipment continues to function consistent with existing accident analyses.

3. Does the proposed activity involve a significant reduction in a margin of safety defined in the basis of any Operating License Technical Specification?

Power Uprate will not involve a significant reduction in a margin of safety. The bounding events which had been analyzed in the UFSAR were reevaluated to demonstrate that power uprate can be implemented without exceeding any analyzed limit. Because the applicable safety analysis criteria and limits are satisfied for power uprate, the margin of safety associated with the safety limits and other limits identified in the Technical Specifications will be maintained.

As discussed in Section 5 of GE Nuclear Energy's License Topical Report NEDO-31984P ''Generic Evaluations of General Electric Boiling Water Reactor Power Uprate,'' the safety margins prescribed by the Code of Federal Regulations (CFR) have been maintained by meeting the appropriate regulatory criteria. Similarly, the margins provided by the application of the ASME design criteria have been maintained. The Brunswick unique analysis NEDC-32466P ''Power Uprate Safety Analysis Report for Brunswick Steam Electric Plant Units 1 and 2'' discusses the effects of power uprate on safety margins for (1) fuel thermal limits, (2) design basis accidents and the challenges for fuel, containment and radiological releases, (3) transient analysis, (4) non-LOCA radiological releases, and (5) environmental consequences. These evaluations conclude that applicable safety analysis criteria and limits are satisfied, and thus, the margins of safety will be maintained.

The changes to the Technical Specification instrumentation will not involve a reduction in the margin of safety. The calculations performed for power uprate have established an analytical limit and calculated the TS Allowable Value and Nominal Trip Setpoint using formal setpoint methodology. This ensures the instrumentation functional requirements are met.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: University of North Carolina at Wilmington, William Madison Randall Library, 601 S. College Road, Wilmington, North Carolina 28403-3297.

Attorney for licensee: William D. Johnson, Vice President and Senior Counsel, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602

NRC Project Director: Eugene V. Imbro

Carolina Power & Light Company, Docket No. 50-261, H. B. Robinson Steam Electric Plant, Unit No. 2, Darlington County, South Carolina

Date of amendment request: March 29, 1996

Description of amendment request: The proposed amendment would revise the technical specifications (TS) to add an allowance to complete a TS required surveillance within 24 hours of discovery of a missed surveillance in accordance with the guidance of Generic Letter (GL) 87-09, ''Sections 3.0 and 4.0 of the Standard Technical Specifications (STS) on the Applicability of Limiting Conditions for Operation and Surveillance Requirements.'' The wording specifying intervals for testing has been changed to reflect wording consistent the new STS. Typographical errors in the basis are also being corrected.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes clarify and incorporates [sic] NRC guidance for application of extending or moving surveillance intervals by plus or minus 25%, by elimination of restrictive surveillance interval descriptions that conflict with NRC guidance, by allowing for an additional 24 hours to perform missed surveillances, and by providing a defined finite period for the term ''immediate'' for Technical Specification (TS) and Inservice Inspection (ISI) surveillances. The basis for extending or moving surveillances, as stated in GL 89-14, ''Line-Item Improvements in Technical Specifications - Removal of the 3.25 Limit on Extending Surveillance Intervals,'' is to provide plants flexibility for scheduling the performance of surveillances and to permit consideration of plant operating conditions that may not be suitable for conducting a surveillance at the specified time interval. Such operating conditions include transient plant operation or ongoing surveillance or maintenance activities. Extending surveillance intervals during plant operation can result in a benefit to safety when a scheduled surveillances [sic] is due at a time that is not suitable for conducting the scheduled surveillance. NUREG-1431, ''Standard Technical Specifications - Westinghouse Plants,'' states ''the 25% extension does not significantly degrade the reliability that results from performing the surveillance at its specified frequency.'' This is based on the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the surveillance requirements. The basis for the 24 hour delay period, as stated in the basis for NUREG-1431, includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the surveillance, the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the requirements.'' The basis for defining the term ''immediate'' is to provide guidance to plant personnel for conducting operability testing of the Steam Driven Auxiliary Feedwater pump after extended shutdown periods in order to minimize plant risks and not pose an unsafe operational transient during an unstable plant configuration (i.e., during plant startup). Since these changes do not affect plant design, operation, or the manner in which testing is performed, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes clarify and incorporates [sic] NRC guidance for application of extending or moving surveillance intervals by plus or minus 25%, by elimination of restrictive surveillance interval descriptions that conflict with NRC guidance, by allowing for an additional 24 hours to perform missed surveillances, and by providing a defined finite period for the term ''immediate'' for TS and ISI surveillances. Since these changes do not affect plant design, operation, or the manner in which testing is performed, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed changes do not involve a significant reduction in the margin of safety.

The changes proposed, with the exception of allowing an additional 24 hours to complete missed surveillances, are to clarify existing surveillance intervals and to provide more specific and detailed criteria without changing current surveillance scheduling methodologies. The NRC has determined that allowing an additional 24 hours to complete missed surveillance tests minimizes additional challenges to plant operations such that there is a conservative balance between the risk associated with performing the surveillance during stable plant conditions and the risk of imposing a plant transient due to TS action statements or changing ''modes'' of operation. These extensions are current industry practices endorsed by the NRC which provide flexibility for scheduling and performing surveillances and permit consideration of plant operating conditions that may not be suitable for conducting a surveillance at either the specified time interval or inadvertently missing the surveillance interval. The risk to safety is low in contrast to the alternatives; therefore, the proposed changes do not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Hartsville Memorial Library, 147 West College Avenue, Hartsville, South Carolina 29550

Attorney for licensee: William D. Johnson, Vice President and Senior Counsel, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602

NRC Project Director: Eugene V. Imbro

Commonwealth Edison Company, Docket Nos. 50-373 and 50-374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of amendment request: April 8, 1996

Description of amendment request: The proposed amendments would change various sections of the Technical Specifications (TS) to reflect the transition of fuel supplier from Generic Electric to Siemens Power Corporation (SPC). The amendments would revise the definitions and Limiting Conditions for Operation related to Linear Heat Generation Rate, Critical Power Ratio, Maximum Critical Power Ratio, and Fraction of Limiting Power Density to incorporate SPC terms and methodology or to make the TS vendor neutral. Section 6.0 of the TS would be revised to include SPC references. The proposed amendment also adds a requirement to adjust the Average Planar Linear Heat Generation Rate when the reactor is in single loop operation since SPC methodologies may require this reduction factor for SPC fuel. The SPC methodologies to be added to the TS have previously been approved by the NRC. The proposed amendment would also relocate requirements for the traversing in-core probe system from the TS to the Core Operating Limits Report and would upgrade the fuel description in Section 5.0 as a line item from the Improved Technical Specifications.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

The probability of an evaluated accident is derived from the probabilities of the individual precursors to that accident. The consequences of an evaluated accident are determined by the operability of plant systems designed to mitigate those consequences. Limits will be established consistent with NRC approved methods to ensure that fuel performance during normal, transient, and accident conditions is acceptable. The proposed Technical Specifications amendment reflects previously approved SPC methodology used to analyze normal operations, including anticipated operational occurrences (AOOs), and to determine the potential consequences of accidents.

Licensing Methods and Models

The proposed amendment is to support operation with NRC approved fuel and licensing methods supplied from Siemens Power Corporation. In accordance with FSAR Chapter 15, the same accidents and transients will be analyzed with the new fuel and methods as were analyzed by GE for GE fuel. The analysis methods and models are NRC approved (Note the mixed core treatment of critical power ratio is being addressed under separate correspondence). These approved methods and models are used to determine the fuel thermal limits. Traversing In-core Probe (TIP) uncertainty are assumptions in the approved Siemens core monitoring methodologies. The SPC core monitoring code enables the site to monitor k_{eff} as well as rod density to perform the reactivity anomaly surveillance. This is consistent with GE methodology. Therefore, the change in licensing analysis methods and models does not significantly increase the probability of an accident or the consequences of an accident previously identified. The support systems for minimizing the consequences of transients and accidents are not affected by the proposed amendment.

New Fuel Design

The use of ATRIUM 9B fuel at LaSalle does not involve a significant increase in the probability or consequences of any accident previously evaluated in the FSAR. The ATRIUM-9B fuel is generically approved for use as a reload BWR fuel type. (See Boiling Water Reactor Licensing Methodology Summary, Siemens Power Corporation, EMF-94-217(NP)). Limiting postulated occurrences and normal operation have been analyzed using NRC-approved methods for the ATRIUM 9B fuel design to ensure that safety limits are protected and that acceptable transient and accident performance is maintained.

The reload fuel has no adverse impact on the performance of in-core neutron flux instrumentation or control rod drive response. The ATRIUM-9B fuel design will not adversely affect performance of neutron instrumentation nor will it adversely affect the movement of control blades. The exterior dimensions of the ATRIUM-9B fuel assembly are essentially identical to the GE9B; the ATRIUM-9B fuel assembly for LaSalle uses a standard fuel channel and normal control cell positioning (i.e., no offset). Thus, no adverse interactions with the adjacent control blade and nuclear instrumentation are anticipated. Additionally, given the above mentioned overall envelope similarities, no problems are anticipated with other station equipment such as the fuel storage racks, the new fuel inspection stand and the spent fuel pool fuel preparation machine.

The ATRIUM 9B design is neutronically compatible with the existing fuel types and core components in the LaSalle core. SPC tests have demonstrated that the ATRIUM-9B fuel design is hydraulically compatible with the GE9 fuel. The bundle pressure drop characteristics of the ATRIUM 9B bundle are similar to those of the GE9 fuel design, hence core thermal-hydraulic stability characteristics are not adversely affected by the ATRIUM 9B design.

An evaluation of the Emergency Procedures is being performed to ensure that the use of the ATRIUM-9B fuel at LaSalle does not alter any assumptions previously made in evaluating the radiological consequences of an accident at LaSalle Station.

Methods approved by the NRC are being used in the evaluation of fuel performance during normal and abnormal operating conditions. The ComEd and SPC methods to be used for the cycle specific transient analyses have been previously NRC approved. The exception is the mixed core treatment of critical power ratio, which is being addressed under separate correspondence.

The description of the fuel is expanded to be consistent with NUREG-1434. The description of the fuel materials, lead test assembly use, and stating that designs must have been analyzed with NRC Staff approved codes does not change existing methods; it only describes them.

Review of the above concludes that the probability of occurrence and the consequences of an accident previously evaluated in the safety analysis report have not been significantly increased.

* * * * *

2. Create the possibility of a new or different kind of accident from any accident previously evaluated:

Creation of the possibility of a new or different kind of accident would require the creation of one or more new precursors of that accident. New accident precursors may be created by modifications of the plant configuration, including changes in allowable modes of operation.

Licensing Methods and Models

The proposed Technical Specification amendment reflects previously approved SPC methodology used to analyze normal operations, including AOOs, and to determine the potential consequences of accidents. As stated above, the proposed changes do not permit modes of reactor operation which differ from those currently permitted.

New Fuel Design

The basic design concept of a 9x9 fuel pin array with an internal water box has been used in various lead assembly programs and in reload quantities in Europe since 1986. WNP-2 has loaded reload quantities since 1991. Approximately 650 water box assemblies have been irradiated in the United States through 1995, with a substantially higher number being irradiated overseas. The NRC has reviewed and approved the ATRIUM-9B fuel design. (See Boiling Water Reactor Licensing Methodology Summary, Siemens Power Corporation, EMF-94-217(NP)). The similarities in fuel design and operation indicate there would be no expectation of introducing new or different types of accidents than have been considered for the existing fuel. Therefore, the use of ATRIUM-9B fuel at LaSalle does not create the possibility of a new or different kind of accident from any accident previously evaluated.

* * * * *

3. Involve a significant reduction in the margin of safety for the following reasons:

The existing margin to safety is provided by the existing acceptance criteria (e.g., 10CFR50.46 limits). The proposed Technical Specification amendment reflects previously approved SPC methodology used to demonstrate that the existing acceptance criteria are satisfied. The revised methodology has been previously reviewed and approved by the USNRC for application to reload cores of GE BWRs. References for the Licensing Topical Reports which document this methodology, and include the Safety Evaluation Reports prepared by the USNRC, are added to the Reference section of the Technical Specifications as part of this amendment.

Licensing Methods and Models

The proposed amendment does not involve changes to the existing operability criteria. NRC approved methods and established limits (implemented in the Core Operating Limits Report) ensure acceptable margin is maintained. The ComEd and SPC reload methodologies for the ATRIUM-9B reload design are consistent with the Technical Specification Bases. The Limiting Conditions for Operation are taken into consideration while performing the cycle specific and generic reload safety analyses. NRC approved methods are listed in Specification 6.0 of the Technical Specifications.

Analyses performed with NRC-approved methodology have demonstrated that fuel design and licensing criteria will be met during normal and abnormal operating conditions. Therefore, there is not a significant reduction in the margin of safety.

New Fuel Design

The exterior dimensions of the ATRIUM-9B fuel assembly are essentially identical to the GE9B; the ATRIUM-9B fuel assembly for LaSalle uses a standard fuel channel and normal control cell positioning; i.e., no offset. Thus, no adverse interactions with the adjacent control blade and nuclear instrumentation are anticipated. The change does not adversely impact equipment important to safety and, therefore does not reduce the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the requested amendments involve no significant hazards consideration.

Local Public Document Room location: Jacobs Memorial Library, Illinois Valley Community College, Oglesby, Illinois 61348.

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One

First National Plaza, Chicago, Illinois 60603

NRC Project Director: Robert A. Capra

Commonwealth Edison Company, Docket Nos. 50-373 and 50-374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of amendment request: April 9, 1996

Description of amendment request: The proposed amendments would eliminate the automatic reactor scram function and the group 1 and 3 isolation valve closure functions associated with the Main Steam Line Radiation Monitoring (MSLRM) system high radiation setpoint. Elimination of these functions will eliminate potential spurious scrams and isolations caused by increased main steam line radiation levels during hydrogen injection. The licensee also proposes to raise the MSLRM system alarm setpoints which are not part of the Technical Specifications to include increased background radiation during hydrogen injection. The proposed amendment would also delete the surveillance requirements for the associated instruments.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated because:

Redefining the full power radiation background, thus changing the MSLRM alarm setpoint, does not change the probability of occurrence of any accident which has been postulated and analyzed in the UFSAR, but will reduce the probability of the inadvertent MSIV closure transient which is an analyzed transient in the UFSAR. It does not change the probability of malfunction of any equipment important to safety associated with [loss of coolant accident] LOCA, fuel handling accident or [control rod drop accident] CRDA. It also does not change the resultant offsite radiological dose from the bounding design basis CRDA. This is based upon all radioactivity, resulting from the design basis CRDA, going to the condenser instantaneously (or independent of the actual MSLRM setpoint) in the offsite dose calculation.

The elimination of reactor scram and isolation of MSIVs, isolation of main steam line drain valves and reactor water sample line valves, associated with the MSLRM system actuation do not introduce, mitigate, or reduce the probability of any design basis accident, or any accident, evaluated in the UFSAR. The topical report NEDO-31400A has shown that there is essentially no reasonable radiological consequence benefit in a design basis CRDA of retaining the MSLRM associated reactor scram and MSIV isolation function. In addition, the probability of inadvertent scram and isolation is reduced. The proposed change will not adversely impact the operation of the [reactor protection system] RPS or [primary containment isolation system] PCIS with respect to performing its other intended safety functions. The proposed change will not affect the operation of other plant systems or equipment important to safety. The consequences of eliminating the automatic closure of the main steam line drain isolation valves and reactor recirculation water sample line isolation valves along with the MSIVs has been evaluated to be negligible additions to the CRDA doses. A [LaSalle County Station] LSCS unique analysis has demonstrated that the radiological doses as a result of design basis CRDA are acceptable.

The MSLRM system high radiation trip was intended to function in response to a CRDA which has been previously evaluated. No credit for MSIV closure was taken in the CRDA analysis since it postulates that all the radioactive material assumed to be released from the fuel is transported to the main condenser prior to MSIV closure. Furthermore, the probability of a fuel failure is independent of the operation of the MSLRM system.

By eliminating the MSLRM induced MSIV closure, the Offgas system can be utilized to reduce potential offsite doses after a CRDA. The [mechanical vacuum pump] MVP is tripped no later than 15 minutes of a Hi-Hi radiation alarm but analytically results in acceptable offsite doses.

Thus the proposed amendment will not increase the probability of any accident previously evaluated, and the elimination of the MSLRM isolation signal for MSIVs and other small containment valves will not significantly increase the consequences of a CRDA as previously evaluated.

2) Create the possibility of a new or different kind of accident from any accident previously evaluated because:

Redefining the full power radiation background, thus changing the actual MSLRM alarm setpoint, does not alter the configuration of the plant. It does not revise any logic or function of the MSLRM trip channels or add, replace, or delete any equipment important to safety. Therefore it does not introduce any new failure modes or create any possibility of a new accident which may challenge safety to the public and has not been previously analyzed. It also does not involve any equipment which either has not been evaluated previously, or may have any safety consequences to the public.

The proposed Technical Specification changes involve eliminating the MSLRM system high radiation trip function for initiating an automatic reactor scram, and automatic isolations. The proposed changes will not affect the operation of other plant systems or equipment important to safety. The MSLRM system will continue to initiate alarms as before. Plant procedures will be in place to take appropriate mitigative measures in response to a high alarm.

The isolation and reactor scram functions associated with the MSLRM system actuation were originally intended to mitigate, not prevent, a potential accident scenario such as a CRDA or gross fuel failure event. Adding or removing an electronic signal, such as the one from the MSLRM system, does not change system or hardware design within the reactor vessel pressure boundary, and therefore will not create the possibility of a new or different kind of accident from those evaluated in the UFSAR like a LOCA or CRDA during power operation. It also does not create the possibility of a new or different kind of accident outside the reactor vessel pressure boundary from those evaluated in the UFSAR, such as a LOCA or Fuel Handling Accident. Removing the isolation signal also reduces the probability of inadvertent scram and isolation.

Therefore the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously analyzed.

3) Involve a significant reduction in the margin of safety because:

The current MSLRM trip Hi-Hi alarm setpoint (about 4 R/hour with full power background at 1.3 R/hour) is at 3 times the full power radiation background. As indicated in the plant unique analytical result for LSCS, the radiological reading at the MSLRMs for design basis CRDA is equivalent to over 1200 times the normal full power radiation background (1600 R/hour divided by 1.3 R/hour), or 150 times the full power radiation background during peak HWC environment (since the radiation background is 8 times the normal background). Thus the safety margin was very large, and would still be quite large with the HWC background factored into the MSLRM actuation setpoint (3 x 8 x 1.3 = about 50). The Hi alarm setpoint of 1.5 times full power background likewise will have a higher safety margin. Thus there is basically no adverse consequence to the margin of safety in the basis for the LaSalle technical specifications.

The proposed Technical Specification changes to eliminate the MSLRM system high radiation trip function for initiating an automatic reactor scram, and automatic closure of the MSIVs, main steam line drain isolation valves, and reactor recirculation water sample line isolation valves do not cause radiological dose consequences to exceed the limit established by SRP 15.4.9.

Per NEDO-31400A, the elimination of MSLRM trip/scram signal will result in the reduction of potential inadvertent scrams, unnecessary safety-related actuations, undue vessel isolation, and duty challenges during normal plant operation. These can be interpreted to be a potential reduction in core damage frequency, which translates to an improvement in the margin of safety.

Thus the margin of safety as defined in the basis of the technical specifications is essentially unaffected, and is therefore acceptable.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the requested amendments involve no significant hazards consideration.

Local Public Document Room location: Jacobs Memorial Library, Illinois Valley Community College, Oglesby, Illinois 61348.

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60603

NRC Project Director: Robert A. Capra

Commonwealth Edison Company, Docket Nos. 50-373 and 50-374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of amendment request: April 16, 1996

Description of amendment request: The proposed amendments would eliminate the Technical Specification requirement to perform response time testing for selected instruments. The instruments affected are the sensors for selected reactor protection system instrumentation, main steam isolation actuation instrumentation, and all sensors for emergency core cooling system (ECCS) actuation instrumentation. The proposed changes are supported by analyses performed by the Boiling Water Reactor Owners' Group as documented in NEDO-32291-A which was approved by the NRC for use in license amendment applications.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated because:

The purpose of the proposed Technical Specification (TS) change is to eliminate response time testing requirements for selected components in the Reactor Protection System (RPS), Isolation Actuation instrumentation and Emergency Core Cooling System (ECCS) actuation instrumentation. The Boiling Water Reactor Owners' Group (BWROG) has completed an evaluation which demonstrates that response time testing is redundant to the other TS-required testing. These other tests, in conjunction with actions taken in response to NRC Bulletin 90-01, ''Loss of Fill-Oil in Transmitters Manufactured by Rosemount,'' and Supplement 1, are sufficient to identify failure modes or degradations in instrument response time and ensure operation of the associated systems within acceptable limits. There are no known failure modes that can be detected by response time testing that cannot also be detected by the other TS-required testing. This evaluation was documented in NEDO-32291-A, ''System Analyses for the Elimination of Selected Response Time Testing Requirements,'' dated October 1995. LaSalle County Station, LaSalle, has confirmed the applicability of this evaluation to LaSalle. In addition, LaSalle will complete the actions identified in the NRC staffs safety evaluation of NEDO-32291-A.

Because of the continued application of other existing TS-required tests such as channel calibrations, channel checks, channel functional tests, and logic system functional tests, the response time of these systems will be maintained within the acceptance limits assumed in plant safety analyses and required for successful mitigation of an initiating event. The proposed changes do not affect the capability of the associated systems to perform their intended function within their required response time, nor do the proposed changes themselves affect the operation of any equipment. As a result, LaSalle has concluded that the proposed changes do not involve a significant increase in the probability or the consequences of an accident previously evaluated.

2) Create the possibility of a new or different kind of accident from any accident previously evaluated because:

The proposed changes only apply to the testing requirements for the components identified above and do not result in any physical change to these or other components or their operation. As a result no new failure modes are introduced. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3) Involve a significant reduction in the margin of safety because:

The current TS-required response times are based on the maximum allowable values assumed in the plant safety analyses. These analyses conservatively establish the margin of safety. As described above, the proposed changes do not affect the capability of the associated systems to perform their intended function within the allowed response time used as the basis for the plant safety analyses. The potential failure modes for the components within the scope of this request were evaluated for impact on instrument response time. This evaluation confirmed that, with the exception of loss of fill-oil of Rosemount transmitters, the remaining TS-required testing is sufficient to identify failure modes or degradations in instrument response times and ensure that operation of the applicable instrumentation is within acceptable limits. The actions taken in response to NRC Bulletin 90-01 and Supplement 1 are adequate to identify loss of fill-oil failures of Rosemount transmitters. As a result, it has been concluded that plant and system response to an initiating event will remain in compliance with the assumptions of the safety analysis.

Further, although not explicitly evaluated, the proposed changes will provide an improvement to plant safety and operation by the following:

a. Reducing the time safety systems are unavailable,

b. Reducing the potential for safety system actuations,

c. Reducing plant shutdown risk,

d. Limiting radiation exposure to plant personnel, and

e. Eliminating the diversion of key personnel resources to conduct unnecessary testing.

Therefore, LaSalle has concluded that this request will not significantly reduce the margin of safety, and may actually cause an increase in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the requested amendments involve no significant hazards consideration.

Local Public Document Room location: Jacobs Memorial Library, Illinois Valley Community College, Oglesby, Illinois 61348.

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60603

NRC Project Director: Robert A. Capra

Duke Power Company, et al., Docket Nos. 50-413 and 50-414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of amendment request: November 2, 1994

Description of amendment request: The proposed amendments would delete the content of Appendix B, ''Environmental Protection Plan'' (nonradiological), and modify License Condition 2.C.(2) to delete that portion which refers to the Environmental Protection Plan.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. [The proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated]:

Deletion of the Environmental Protection Plan and modifying License Condition 2.C.(2) will have no impact on the probability or consequences of an accident previously evaluated because the changes will not have any impact upon the design or operation of any plant systems or components.

2. [The proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated]:

The proposed revision will not create the possibility of a new or different kind of accident from any previously evaluated because the revision is administrative in nature and will not change the types and amounts of effluent that will be released.

3. [The proposed amendments would not involve a significant reduction in a margin of safety]:

The proposed revision will not reduce a margin of safety because it is administrative in nature and will not [a]ffect the margin of safety as defined in the basis for any Technical Specifications.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: York County Library, 138 East Black Street, Rock Hill, South Carolina 29730

Attorney for licensee: Mr. Albert Carr, Duke Power Company, 422 South Church Street, Charlotte, North Carolina 28242

NRC Project Director: Herbert N. Berkow

Duquesne Light Company, et al., Docket No. 50-412, Beaver Valley Power Station, Unit 2, Shippingport, Pennsylvania

Date of amendment request: April 29, 1996

Description of amendment request: The proposed amendment would revise Technical Specification (TS) 5.3.1 to allow the use of ZIRCO as an alternate zirconium-based fuel rod material and remove the word clad since it has been eliminated from the text of the NRC's improved Standard Technical Specifications (NUREG-1431). Limited substitution of fuel rods by ZIRCO filler rods would also be permitted. The proposed amendment would revise Note 2 on TS Table 3.9-1 to specify that the maximum burnup in the peak fuel rod in a fuel assembly stored in Region 2 spent fuel racks should not exceed the NRC-approved limit for WCAP-12610 rather than the current maximum burnup limit of 60 GWD/MTU.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

The methodologies used in the accident analyses remain unchanged. The proposed changes do not change or alter the design assumptions for the systems or components used to mitigate the consequences of an accident. Use of ZIRLO fuel rod material does not adversely affect fuel performance or impact nuclear design methodology. Therefore, accident analysis results are not impacted.

The operating limits will not be changed and the analysis methods to demonstrate operation within the limits will remain in accordance with NRC approved methodologies. Other than the changes to the fuel assemblies, there are no physical changes to the plant associated with this technical specification change. A safety analysis will continue to be performed for each cycle to demonstrate compliance with all fuel safety design bases.

VANTAGE 5 fuel assemblies with ZIRLO fuel rods meet the same fuel assembly and fuel rod design bases as other VANTAGE 5 fuel assemblies. In addition, the 10 CFR 50.46 criteria are applied to the ZIRLO fuel rods. The use of these fuel assemblies will not result in a change to the reload design and safety analysis limits. Since the original design criteria are met, the ZIRLO fuel rods will not be an initiator for any new accident. The fuel rod material is similar in chemical composition and has similar physical and mechanical properties as Zircaloy-4. Thus, the fuel rod integrity is maintained and the structural integrity of the fuel assembly is not affected. ZIRLO improves corrosion performance and dimensional stability. No concerns have been identified with respect to the use of an assembly containing a combination of Zircaloy-4 and ZIRLO fuel rods.

The dose predictions in the safety analyses are not sensitive to the fuel rod material used; therefore, the radiological consequences of accidents previously evaluated in the safety analysis remain valid. A reload analysis is completed for each cycle, in accordance with NRC approved methodologies. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated.

VANTAGE 5 fuel assemblies with ZIRLO fuel rods satisfy the same design bases as those used for other VANTAGE 5 fuel assemblies. All design and performance criteria continue to be met and no new failure mechanisms have been identified. The ZIRLO fuel rod material offers improved corrosion resistance and structural integrity.

The proposed changes do not affect the design or operation of any system or component in the plant. The safety functions of the related structures, systems, or components are not changed in any manner, nor is the reliability of any structure, system, or component reduced. The changes do not affect the manner by which the facility is operated and do not change any facility design feature, structure, or system. No new or different type of equipment will be installed. Since there is no change to the facility or operating procedures, and the safety functions and reliability of structures, systems, or components are not affected, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

The use of Zircaloy-4, ZIRLO, or stainless steal filler rods in fuel assemblies will not involve a significant reduction in the margin of safety because analyses using NRC approved methodology will be performed for each configuration to demonstrate continued operation within the limits that assure acceptable plant response to accidents and transients. These analyses will be performed using NRC approved methods that have been approved for application to the fuel configuration.

Use of ZIRLO as fuel rod material does not change the VANTAGE 5 reload design and safety analysis limits. The use of these fuel assemblies will take into consideration the normal core operating conditions allowed in the technical specifications. For each reload core, the fuel assemblies will be evaluated using NRC approved reload design methods, including consideration of the core physics analysis peaking factors and core average linear heat rate effects.

Based on the above, it is concluded that the proposed license amendment request does not result in a significant reduction in margin with respect to plant safety as defined in the UFSAR [Updated Final Safety Analysis Report] or any plant technical specification BASES.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: B. F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, Pennsylvania 1500l.

Attorney for licensee: Jay E. Silberg, Esquire, Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: John F. Stolz

Entergy Operations, Inc., Docket Nos. 50-313 and 50-368, Arkansas Nuclear One, Unit Nos. 1 and 2 (ANO-1&2), Pope County, Arkansas

Date of amendment request: May 2, 1996

Description of amendment request: The proposed technical specification amendments would extend the allowed outage times for emergency diesel generators at Arkansas Nuclear One, Units 1 and 2 to 7 days with an additional, once per refueling cycle extension of 7 more days for each machine.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Criterion 1 - Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The emergency diesel generators (EDGs) are backup alternating current power sources designed to power essential safety systems in the event of a loss of offsite power. The EDGs are not accident initiators in any accident previously evaluated. Probabilistic Safety Analysis (PSA) methods were utilized in order to fully evaluate the EDG allowed outage time (AOT) extension proposed in this submittal. The results of these analyses indicate there is not a significant increase in the probability of an accident previously evaluated. Therefore, this change does not involve an increase in the probability of an accident previously evaluated.

The EDGs provide backup power to components that mitigate the consequences of accidents. The current TSs allow for an EDG to be removed from service for an AOT. The proposed amendment extends the current AOT for an EDG. The proposed change does not allow any more equipment to be removed from service at one time. The proposed changes to the AOTs do not affect any of the assumptions used in deterministic safety analysis. By extending the EDG AOT, the consequences of an accident previously evaluated will remain unchanged.

The proposed change removes redundant requirements associated with an inoperable emergency power supply from the TS for the pressurizer proportional heaters. The operability requirements for emergency power supplies and actions to be taken if an EDG is inoperable are already addressed in the ANO-2 TS 3.8.1.1.

The associated changes that remove the requirements to test the EDGs if one or both offsite power supplies are inoperable, for an inoperable station battery, for an inoperable component in the two ESF electrical distribution systems, the accelerated testing requirements of the EDGs, and the daily testing requirements for the operable EDGs improve the reliability for the operable EDGs by reducing the number of unnecessary starts and stops. By improving the EDG reliability, this change will not increase the consequences of the accidents previously evaluated.

The other changes in this submittal associated with the bases are considered administrative in nature and have no effect on the consequences of an accident previously evaluated.

Therefore, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

Criterion 2 - Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

This proposed change does not alter the design, configuration, or method of operation of the plant. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

Criterion 3 - Does Not Involve a Significant Reduction in the Margin of Safety.

The proposed changes do not affect the Technical Specification limiting conditions for operation or their bases which support the deterministic analyses used to establish the margin of safety.

Calculations performed to analyze the change in risk based on these changes produced acceptable values which are included in the tables located in the description of changes section. These calculated changes in risk fall well within that which is normally considered acceptable. When the additional benefit of maintaining the Emergency Diesel Generators available during shutdown cooling operations associated with refueling outages in considered, the overall change in risk is further reduced.

The remaining proposed changes are either associated with increasing EDG reliability or considered administrative in nature.

Therefore, this change does not involve a significant reduction in the margin of safety

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Tomlinson Library, Arkansas Tech University, Russellville, AR 72801

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., Washington, DC 20005-3502

NRC Project Director: William D. Beckner

Entergy Operations, Inc., et al., Docket No. 50-416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of amendment request: November 20, 1995, as supplemented by the letter dated December 15, 1995.

Description of amendment request: The licensee has proposed to revise the Grand Gulf Nuclear Station (GGNS), Unit 1, Technical Specifications (TSs) as follows for the drywell, the drywell airlock, and the drywell isolation valves:

1. For the drywell in Limiting Condition of Operation (LCO) 3.6.5.1, the surveillance frequency interval for the drywell bypass test in Surveillance Requirement (SR) 3.6.5.1.1 would be increased from 18 months to 10 years. For this interval change, an increased testing frequency would be required if bypass performance degrades (i.e., the leakage is greater than the limit for two consecutive tests) and the application of SR 3.0.2, the allowance to extend the surveillance interval by 25 percent, would be restricted to 12 months on the 10-year interval. This includes deleting the Note in SR 3.6.5.1.1.

2. For the drywell airlock in LCO 3.6.5.2, the following changes are requested: (a) the leak rate SR 3.6.5.2.2 would be transferred from the airlock LCO (3.6.5.2) to SR 3.6.5.1.3 in the drywell LCO (3.6.5.1), (b) the requirement in SR 3.6.5.2.2 for the air lock to meet a specific overall leakage limit would be deleted, (c) the Note in SR 3.6.5.2.2 that stated that an inoperable air lock door does not invalidate the previous air lock leakage test would be deleted, (d) the test pressure for the air lock leakage test in SR 3.6.5.2.2 would be reduced from 11.5 psig to 3 psid, and (e) the surveillance frequency interval for the air lock leakage and interlock testing, required in SRs 3.6.5.2.1 and 3.6.5.2.2, would be increased from 18 months to 24 months.

3. For the drywell airlock in LCO 3.6.5.2 and the drywell isolation valves in LCO 3.6.5.3, the Action Notes, which identify that the actions required by drywell LCO 3.6.5.1 must be taken when the drywell bypass leakage limit is not met, would be deleted. Action C.1 of LCO 3.6.5.2 and its associated completion time would also be deleted.There would also be changes to the Bases of the TSs for the above LCOs and SRs, based on the proposed changes.

Basis for proposed no significant hazards consideration determination: The amendment request dated November 20, 1995, applied to both the Grand Gulf Nuclear Station (GGNS) and the River Bend Station (RSB); however, not all of the proposed amendments apply to GGNS. This Notice only discusses the amendment request for GGNS. The reference below to proposed amendments which do not apply to GGNS are marked by ''[....]''.

As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration in its application dated November 20, 1995, which is presented below:

Entergy Operations, Inc. proposes to change the current Grand Gulf Nuclear Station (GGNS) [....] Technical Specifications. The specific proposed changes are:

1. The Surveillance Frequency [interval] for the drywell bypass test is changed [increased] from 18 months to 10 years with an increased testing frequency required if performance degrades.

2. The following changes are requested for the drywell air lock testing: (a) the leakage rate surveillance is moved from the air lock Limiting Condition for Operation (LCO) to the drywell LCO, (b) the requirement for the air lock to meet a specific overall leakage limit is deleted, (c) the Note that an inoperable air lock door does not invalidate the previous air lock leakage test is deleted, (d) the GGNS test pressure for the air lock leakage test is changed [reduced] from 11.5 psig to 3 psid, [...,] and ([e]) the Surveillance Frequency [interval] for the air lock leakage test and interlock test is changed [increased] from 18 months to 24 months.

3. The Actions Notes in the drywell air lock LCO and the drywell isolation valve LCO that identifies that the Actions required

by the drywell LCO must be taken when the drywell bypass leakage limit is not met is deleted. [Action C.1 of LCO 3.6.5.2 and its associated completion time would also be deleted.]

[4. ...]

The Commission has provided standards for determining whether a no significant hazards consideration exists as stated in 10 CFR 50.92(c). The proposed changes involve the withdrawal of operating restrictions previously imposed because acceptable operation of the Mark III primary containment design had not been demonstrated at the time of licensing. As published in the Federal Register regarding no significant hazards consideration criteria, granting of a relief, based upon demonstration of acceptable operation from an operating restriction that was imposed because acceptable operation had not yet been demonstrated does not involve a significant hazards consideration (Ref. 48 FR 14870). Furthermore, a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

Entergy Operations, Inc. has evaluated the no significant hazards consideration in its request for this license amendment, even though the above-mentioned criterion is satisfied by this proposal. In accordance with 10 CFR 50.91(a), Entergy Operations, Inc. is providing the analysis of the proposed amendment against the three standards in 10 CFR 50.92(c). A description of the no significant hazards consideration determination follows:

I. The proposed change does not significantly increase the probability or consequences of an accident previously evaluated.

The requested changes are either administrative changes which clarify the format of the requirement or change the requirement to match the design bases of the plant, a change which relocates the requirement to the Technical Specification Bases, or a change in [the] surveillance interval. Each of these types of change are discussed below:

1. The administrative changes clarify the format of the requirement or change the requirement to match the design bases of the plant. Clarifying [the] administrative format of the Technical Specifications does not result in any changes to the Technical Specification requirements and, as a result, does not involve a significant increase in the probability or consequences of an accident previously evaluated. Also, changing the requirements of the Technical Specifications to more closely match the design bases of the plant will continue to assure that the plant will respond as assumed in the accident analyses and, as a result, does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed changes relocate information to the Technical Specification Bases. In the Technical Specifications Bases the relocated information will be maintained in accordance with 10 CFR 50.59 and subject to the change control provisions in Chapter 5 of Technical Specifications. Since any changes to the Technical Specifications Bases will be evaluated per the requirements of 10 CFR 50.59, no increase (significant or insignificant) in the probability or consequences of an accident previously evaluated will be allowed. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

3. The proposed changes in frequency for the drywell bypass leakage and drywell air lock surveillances will continue to ensure that no paths exist through passive drywell boundary components that would permit gross leakage from the drywell to the primary containment air space and result in bypassing the primary containment pressure-suppression feature beyond the design basis limit. The Mark III primary containment system satisfies General Design Criterion 16 of Appendix A to 10 CFR Part 50. Maximum drywell bypass leakage was determined previously by reviewing the full range of postulated primary system break sizes. The limiting case was a primary system small break loss of coolant accident (LOCA) and yielded a design allowable drywell bypass leakage rate limit of approximately 35,000 scfm for GGNS [....]. The Technical Specifications acceptable limit for the bypass leakage following a surveillance is less than 10% of this design basis value. The most recent bypass leakage value was approximately 2.5% for GGNS [....] of the design allowable leakage rate limit for the limiting event. EOI [Entergy Operations, Inc.] is committed to maintaining programmatic and oversight controls that ensure that drywell bypass leakage remains a small fraction of the design allowable leakage limit.

The drywell is typically exposed to essentially 0 psig during normal plant operation and 3 psig during drywell bypass leak rate testing. These pressures are considerably lower than the structural integrity test pressure and are less likely to initiate a crack or cause an existing crack to grow. Visual inspections of the accessible drywell surfaces that have been performed since the structural integrity tests have not revealed the presence of additional cracking or other abnormalities. Therefore, additional cracking of the drywell structure is not expected due to testing or operation and, similar to the justification for the ten year 10 CFR 50 Appendix J Type A test interval, it is not considered credible for the passive drywell structure to begin to leak sufficiently to impact the design drywell bypass leakage limit.

The primary containment's ability to perform its safety function is fairly insensitive to the amount of drywell leakage, thereby providing a margin to loss of the drywell safety function that is not normally available for systems. This insensitivity is demonstrated by the extremely high limiting event design basis allowable leakage for the drywell (e.g., 35,000 scfm for GGNS [....]). The limiting leakage is almost an order of magnitude higher for other events. Additionally, an even higher allowable leakage can be realistically accommodated by the primary containment due to the margins in the containment design. Because of the margins available, it will take valves in multiple penetration flow paths leaking excessively to cause the primary containment to fail as a result of overpressurization, the probability that drywell isolation valve leakage will result in primary containment failure due to excessive drywell leakage is not considered significant and this drywell/primary containment failure mode is not considered credible.

The proposed Technical Specification changes have no significant impact on the GGNS Individual Plant Examination (IPE) [....] conducted per NRC Generic Letter 88-20. The IPEs considered overpressurization failure of primary containment as part of the primary containment performance assessment. Due to the magnitude of acceptable drywell leakage and the extremely low probabilities of achieving such leakage, primary containment failure due to preexisting excessive drywell leakage was considered a non significant contributor to primary containment failure. Primary containment overpressurization failure can occur with or without preexisting excessive drywell leakage in a severe accident. This is due to physical phenomena associated with potentially extreme environmental conditions inside primary containment following a severe accident. However, the calculated frequency of such extreme conditions is very small. The proposed changes do not impact the IPE evaluated phenomena causing primary containment overpressurization failure nor significantly increase the probability that the drywell has preexisting excessive leakage and therefore would not contribute to these accident scenarios.

For the reasons discussed above, the proposed changes do not have any significant risk impact to accidents previously evaluated and do not significantly increase the consequences of an accident previously evaluated. Additionally, drywell leakage is not the initiator of any accident evaluated; therefore, changes in the frequency of the surveillance for drywell leakage does not increase the probability of any accident evaluated.

Therefore, the proposed changes do not significantly increase the probability or consequences of an accident previously evaluated.

II. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The requested changes are either administrative changes which clarify the format of the requirement or change the requirement to match the design bases of the plant, a change which relocates the requirement to the Technical Specification Bases, or a change in surveillance interval. Each of these types of change are discussed below:

1. The administrative changes in the Technical Specification requirements do not

involve a physical alteration of the plant (no new or different type of equipment will be installed) nor does it change the methods governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

2. The proposed relocation of requirements does not involve a physical alteration of the plant (no new or different type of equipment will be installed) nor does it change the methods governing normal plant operation. The proposed change will not impose or eliminate any requirements. Adequate control of the information will be maintained in the Technical Specification Bases. Thus, the change proposed does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change modifies the surveillance frequency for drywell bypass leakage and drywell air lock surveillances. The changes only impact the test frequency and do not result in any change in the response of the equipment to an accident. The changes do not alter equipment design or capabilities. The changes do not present any new or additional failure mechanisms. The drywell is passive in nature and the surveillance will continue to verify that its integrity has not deteriorated. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

III. The proposed change does not involve a significant reduction in a margin of safety.

The requested changes are either administrative changes which clarify the format of the requirement or change the requirement to match the design bases of the plant, a change which relocates the requirement to the Technical Specification Bases, or a change in surveillance interval. Each of these types of changes are discussed below:

1. The administrative changes in the Technical Specification requirements do not involve a physical alteration of the plant (no new or different type of equipment will be installed) nor does it change the methods governing normal plant operation. Thus, this change does not cause a significant reduction in the margin of safety.

2. The relocation of requirements will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be transferred from the Technical Specifications to the Technical Specifications Bases are the same as the existing Technical Specifications. Since any future changes to these requirements in the Technical Specifications Bases will be evaluated per the requirements of 10 CFR 50.59, no reduction (significant or insignificant) in a margin of safety will be allowed.

3. The proposed change modifies the surveillance frequency for drywell bypass leakage and associated air lock surveillances. Reliability of drywell integrity is evidenced by the measured leakage rate during past drywell bypass leakage surveillances. Appropriate design basis assumptions will be upheld, even when combined with the complementary bypass leakage surveillances as proposed. Drywell integrity will continue to be tested by means of the proposed periodic drywell bypass leakage test, performance of the drywell air lock door latching and interlock mechanism surveillance, and performance of additional surveillances including exercising of drywell isolation valves. The combination of these surveillances will provide adequate assurance that drywell bypass leakage will not exceed the design basis limit. Margins of safety would not be reduced unless leakage rates exceeded the design allowable drywell bypass leakage limit. Therefore, the proposed change does not cause a significant reduction in the margin of safety.

Therefore, the proposed changes do not cause a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee’s analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Judge George W. Armstrong Library, 220 S. Commerce Street, Natchez, MS 39120

Attorney for licensee: Nicholas S. Reynolds, Esquire, Winston and Strawn, 1400 L Street, N.W., 12th Floor, Washington, DC 20005-3502

NRC Project Director: William D. Beckner

Entergy Operations Inc., Docket No. 50-382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: August 11, 1995, as supplemented by letter dated February 12, 1996.

Description of amendment request: The proposed change will reduce the minimum reactor coolant cold leg temperature from 544 Degrees F to 541 degrees F in Technical Specification Section 3.2.6, ‘‘Reactor Coolant Cold Leg Temperature.’’

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

The proposed change involves a 3°F reduction in the minimum core inlet temperature. This change will not have any impact on the probability of occurrence of any accident documented in the FSAR.

The impact of this change on the consequences of events documented in the FSAR has been evaluated. The evaluation demonstrated that most events are insensitive to the core inlet temperature. The events that are impacted by lower core inlet temperature are:

Loss of condenser vacuum (LOCV),
Part length CEA drop,
Single CEA withdrawal within deadband, and
CEA ejection.

The LOCV event has been reanalyzed for the upcoming Cycle (Cycle 8) and the results indicate that the peak RCS pressure remains below the acceptable limit (110% of the design pressure, i.e., 2750 psia). The reactivity anomaly events (remaining events) will be reanalyzed as part of COLSS/CPC setpoint calculations. These calculations will be performed prior to Cycle 8 startup and will address the impact of the 3°F reduction on the minimum core inlet temperature. The CPC/COLSS databases and/or addressable constants will be modified, as needed due to proposed change, prior to cycle startup.

A qualitative assessment of the impact of the proposed change on the calculated LOCA blowdown loads that are applied to the major NSSS components, their supports and the reactor vessel internals was also performed. This assessment consisted of an evaluation of the design margins on the major components and a determination of the impact this lower temperature would have on those margins. The evaluation concluded that the impact of a 3°F cold leg temperature reduction will be well within the current design margins. Therefore, the proposed change will not involve a significant increase in the probability or consequences of any accident previously evaluated.

The proposed change to the minimum core inlet temperature does not involve any change to any equipment or the manner in which the plant will be operated. Since no hardware modifications or changes in operation procedures will be made, the proposed change would not create the possibility of a new or different kind of accident from any accident previously evaluated. Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

The impact of the proposed change on the Waterford 3 FSAR analyses have been evaluated. The evaluation showed that the events that were impacted were important with respect to RCS pressure and fuel thermal limits. One of the events that was impacted by the proposed change was the LOCV event. This event was analyzed and the results showed that the peak RCS pressure remained below the acceptable limit. The impact of this change on other events (reactivity anomaly events) will be evaluated as part of the COLSS/CPC setpoint calculations and the COLSS/CPC databases and/or addressable constants will modified as needed to account for any adverse impact on the results of these events due to the proposed change.

The impact of this change on the Linear Heat Generation Rate limits which varies as a function of the cold leg temperature, is accounted for by Technical Specification 3.2.1, ‘‘Linear Heat Rate’’. The impact of this change on LOCA blowdown loads were evaluated to be insignificant compared to the current design margins. Therefore, the proposed change will not involve a significant reduction in a margin of safety, specifically fuel thermal limits and RCS pressure limit.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: University of New Orleans Library, Louisiana Collection, Lakefront, New Orleans, LA 70122

Attorney for licensee: N.S. Reynolds, Esq., Winston & Strawn 1400 L Street N.W., Washington, D.C. 20005-3502

NRC Project Director: William D. Beckner

Florida Power and Light Company, Docket Nos. 50-250 and 50-251, Turkey Point Plant Units 3 and 4, Dade County, Florida

Dates of amendment request: March 20, 1996, and April 23, 1996

Description of amendment request: The licensee proposed to change the Turkey Point Units 3 and 4 Technical Specifications (TS) to relocate the requirements for surveillance testing of the water level and pressure channel instrumentation for the reactor coolant system accumulators and clarify the remaining TS surveillance tests. These amendments also modify the existing action statements of TS 3.5.1 for accumulators to reflect the requirements of NUREG-1431 by requiring a 72-hour period to restore boron concentration if it is not within the limits, and a 1-hour period to restore any other condition rendering the accumulators inoperable.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below.

(1) Operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated because the proposed amendments conform to the guidance given in Enclosure 1 of the NRC GL [Generic Letter] 93-05. The overall functional capabilities of the Emergency Core Cooling System (ECCS) accumulators will not be modified by the proposed change. This amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated for the following reasons:

1) The Water Level and Pressure Channel Instrumentation does not perform a specific safety function, and merely provides an indicating function. The instrumentation in no way affects the capability of the accumulators to perform their respective safety function.

2) The changes in most of the ACTION statements are more restrictive than current TS requirements due to the one hour vice four hour completion time, and therefore will not increase the probability or consequences of a previously evaluated accident. If one accumulator is inoperable for a reason other than boron concentration, the accumulator must be returned to OPERABLE status within 1 hour. In this condition, the required contents of three accumulators cannot be assumed to reach the core during a Loss Of Coolant Accident (LOCA). Due to the severity of the consequences should a LOCA occur in these conditions, the 1 hour completion time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action will be taken to return the inoperable accumulator to OPERABLE status. The completion time minimizes the potential for exposure of the plant to a LOCA under these conditions. The 1 hour requirement for restoring a closed isolation valve is merely a clarification of the existing ''immediate'' time requirement.

3) In the case of low-out-of-specification boron concentration in one accumulator, it must be returned to within the limits within 72 hours. In this condition, ability to maintain subcriticality or minimum boron precipitation time may be reduced. The boron in the accumulators contributes to the assumption that the combined ECCS water in the partially recovered core during the early reflooding phase of a large break LOCA is sufficient to keep that portion of the core subcritical. One accumulator below the minimum boron concentration limit, however, will have no effect on available ECCS water and an insignificant effect on core subcriticality during reflood. Boiling of ECCS water in the core during reflood concentrates boron in the saturated liquid that remains in the core. In addition, current Turkey Point analysis demonstrate that the accumulators discharge only a small amount following a large main steam line break. Therefore, their impact on boron concentration in the reactor coolant system is minor and not a design limiting event. Thus, 72 hours is allowed to return the boron concentration to within limits and does not increase the probability or consequences of an accident previously evaluated.

(2) Operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The use of the modified specifications can not create the possibility of a new or different kind of accident from any previously evaluated since the proposed amendments will not change the physical plant or the modes of plant operation defined in the facility operating license. No new failure mode is introduced due to the surveillance changes and clarifications, since the proposed changes do not involve the addition or modification of equipment nor do they alter the design or operation of affected plant systems.

(3) Operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

The operating limits and functional capabilities of the affected system are unchanged by the proposed amendment. The modified specifications which remove surveillance requirements from the TS to plant procedures are consistent with the NRC GL 93-05 line-item improvement guidance do not significantly reduce any of the margins of safety even though the amount of surveillances is decreased. The modification of the existing ACTION Statements do not have an adverse on [sic] affect on the margin of safety for the following reasons:

1) The SI [Safety Injection] Accumulator Water Level and Pressure Channel instrumentation performs no safety function.

2) The changes in ACTION statements a) and b) are for the most part more restrictive than existing TS requirements, the reason being the removal of instrumentation requirements for operability.

3) In the case of low-out-of-specification boron concentration in one accumulator, the requirement will be less restrictive, but the low boron concentration in one accumulator will have no effect on available ECCS water and an insignificant effect on core subcriticality during reflood and therefore will not significantly reduce the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied.Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Florida International University, University Park, Miami, Florida 33199

Attorney for licensee: J. R. Newman, Esquire, Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036

NRC Project Director: Frederick J. Hebdon

Illinois Power Company and Soyland Power Cooperative, Inc., Docket No. 50-461, Clinton Power Station, Unit No. 1, DeWitt County, Illinois

Date of amendment request: April 19, 1996

Description of amendment request: The proposed amendment would include revisions to Technical Specification (TS) 3.3.6.1, ''Primary Containment and Drywell Isolation Instrumentation≥; TS 3.3.6.2, ''Secondary Containment Isolation Instrumentation≥; TS 3.3.7.1, ''Control Room Ventilation System Instrumentation≥; TS 3.6.1.2, ''Primary Containment Air Locks≥; TS 3.6.1.3, ''Primary Containment Isolation Valves≥; TS 3.6.4.1, ''Secondary Containment≥; TS 3.6.4.2, ''Secondary Containment Isolation Dampers≥; TS 3.6.4.3, ''Standby Gas Treatment≥; TS 3.7.3, ''Control Room Ventilation≥; and TS 3.7.4, ''Control Room AC System.'' These TSs would be revised to eliminate CORE ALTERATIONS as an applicable condition for which the associated Limiting Conditions for Operation (LCO) must be met. Consistent changes are also proposed for the associated ACTIONS in each of these LCOs, to reflect the changes in the applicable conditions. The intent of these proposed changes is to allow certain activities such as control rod venting, which is considered a CORE ALTERATION in MODE 5, to be performed without the requirements of the identified LCOs being met.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration which is presented below:

1. The proposed changes eliminate CORE ALTERATIONS as an applicable condition requiring operability of the primary and secondary containment and control room ventilation system. As stated in the BASES for the associated Technical Specifications, operability of these systems is primarily required for mitigation of the design basis accident - fuel handling accident (DBA-FHA) and design basis accident - loss of coolant accident (DBA-LOCA). The performance of CORE ALTERATIONS alone is neither a precursor to, nor a condition during which these DBAs are postulated to occur. The proposed changes only delete CORE ALTERATIONS as an applicable condition for the affected Technical Specifications. All other applicable MODES or specified conditions, including operations with the potential for draining the reactor vessels (OPDRVs) and the movement of irradiated fuel assemblies within the primary or secondary containment, remain unchanged. Further, the limitations placed on the handling of light loads are also unchanged. The Technical Specifications (and the separate requirements imposed on the handling of light loads) will thus continue to require that systems or functions designed to mitigate design-basis/previously evaluated accidents are OPERABLE during the relevant operating MODES or conditions. On the basis of the above, it is concluded that the requested amendment will not increase the probability or consequences of any accident previously evaluated.

2. The proposed changes do not involve any modification to the plant design or to the operation of plant systems (except to determine when certain analyzed accident-mitigating systems or features are required to be OPERABLE). The failure modes considered for the proposed changes are the same as those previously considered, therefore, it can be concluded that no new failure modes will be created. On this basis, the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The changes being made to eliminate CORE ALTERATIONS as an applicable condition for which certain LCOs must be met, do not eliminate the requirements for operability of those systems or features assumed to mitigate design-basis or analyzed accidents during the applicable MODES when such systems or features are assumed to be available for performing their mitigating function. The safety margins assumed or established by the accident analyses for those design-basis events (as described in the accident analyses of the Clinton Power Station Updated Final Safety Analysis Report) therefore remain unchanged. Further, the proposed changes do not impact the controls imposed on the handling of light loads (including unirradiated fuel assemblies) for ensuring that such activities cannot result in an event that yields consequences more severe than those calculated for the DBA-FHA. With respect to reactivity concerns during refueling operations (MODE 5), all systems or features required to be OPERABLE for precluding inadvertent criticality and monitoring reactivity changes will continue to be required OPERABLE as per the current Technical Specification requirements. The deletion of CORE ALTERATIONS as an applicable condition only applies to the noted systems which do not contribute to precluding reactivity events. Based on the above, the proposed changes do not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Vespasian Warner Public Library, 120 West Johnson Street, Clinton, Illinois 61727

Attorney for licensee: Leah Manning Stetener, Vice President, General Counsel, and Corporate Secretary, 500 South 27th Street, Decatur, Illinois 62525

NRC Project Director: Gail H. Marcus

Illinois Power Company and Soyland Power Cooperative, Inc., Docket No. 50-461, Clinton Power Station, Unit No. 1, DeWitt County, Illinois

Date of amendment request: May 1, 1996

Description of amendment request: The proposed amendment would revise the Clinton Power Station (CPS) Operating License and Technical Specifications (TS) to implement 10 CFR Part 50, Appendix J - Option B, by referring to Regulatory Guide 1.163, ''Performance-Based Containment Leak-Test Program.'' Specifically, changes would be made to paragraph 2.D of the Operating License; TS Section 1.1, ''Definitions;'' TS 3.6.1.1, ''Primary Containment;'' TS 3.6.1.1, ''Primary Containment Air Locks;'' TS 3.6.1.3, ''Primary Containment Isolation Valves (PCIVs);'' and TS Section 5.5, ''Programs and Manuals.''

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration which is presented below:

1. The proposed change implements new Option B of 10 CFR 50 Appendix J for performance-based primary containment leakage testing. The proposed change does not involve a change to the plant design or operation. As a result, the proposed change does not affect any parameters or conditions that contribute to the initiation of any accidents previously evaluated. Thus, the proposed change cannot increase the probability of any accident previously evaluated.

The proposed change potentially affects the leak-tight integrity of the primary containment structure which is designed to mitigate the consequences of a loss-of-coolant accident (LOCA) by limiting the release of fission products contained in the post-LOCA primary containment atmosphere. Functional integrity of the primary containment must be maintained during and following the peak transient pressures and temperatures that may result from a LOCA. Because the proposed change does not alter the plant design, including the primary containment and primary containment penetrations, and because it only affects the frequency of measuring Type A, B, and C leakage without changing the acceptance criteria for the Type A, B, and C leakage rate tests, the proposed change does not directly result in an increase in the primary containment leakage. However, decreasing the test frequency can increase the probability that an increase in primary containment leakage could go undetected for an extended period of time. To minimize that probability, test intervals will be established based on the performance history of components being tested.

NUREG-1493, ''Performance-Based Containment Leak-Test Program,'' provides the technical basis for the NRC's rulemaking to revise primary containment leakage testing requirements for nuclear power reactors in 10 CFR 50, Appendix J. NUREG-1493 documents the NRC's determination that the effect of primary containment leakage on overall accident risk is minimal since risk is dominated by accident sequences that result in failure of bypass of primary containment. NUREG-1493 also documents that increasing the Type A leakage test intervals would have a minimal impact on public risk, and that Type B and C tests can identify the vast majority (greater than ninety five percent) of all leakage paths. Therefore, performance-based alternatives to current local leakage-testing requirements are feasible without significant risk impacts.

Based on the above, IP has concluded that the proposed change will not result in a significant increase in the probability or consequences of any accident previously evaluated.

2. The proposed change does not involve a change to the plant design or operation. As a result, the proposed change does not affect any of the parameters or conditions that could contribute to initiation of any accidents. This change involves the reduction of Type A, B, and C test frequency. Except for the method of defining the test frequency, the methods for performing the actual tests are not changed. No new accident modes are created by extending the testing intervals. No safety-related equipment or safety functions are altered as a result of this change. Thus, extending the test frequency has no influence on, nor does it contribute to the possibility of a new or different kind of accident or malfunction from those previously analyzed.

Based on the above, IP has concluded that the proposed change will not create the possibility of a new or different kind of accident not previously evaluated.

3. The request does not involve a significant reduction in a margin to safety. The proposed change only affects the frequency of the Type A, B, and C testing. Except for the method of defining the test frequency, the methods for performing the actual tests are not changed. However, the proposed change can increase the probability that an increase in primary containment leakage could go undetected for an extended period of time. NUREG-1493 has determined that under several different accident scenarios, the increased risk of radioactivity release from primary containment is negligible with the implementation of these proposed changes.

The margin of safety that has the potential of being impacted by the proposed change involves the offsite dose consequences of postulated accidents which are directly related to the rate of primary containment leakage. The primary containment isolation system is designed to limit leakage to L_a, which is defined by the CPS Technical Specifications to be 0.65% of primary containment air weight per day at the calculated peak containment internal pressure for the design basis loss of coolant accident (P_a). The limitation on the rate of primary containment leakage is designed to ensure that the total leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure (P_a). The margin of safety for the offsite dose consequences of postulated accidents directly related to the primary containment leakage rate is maintained by continuing to meet the 1.0 L_a acceptance criteria. The L_a value is not being modified by this proposed change.

Except for the method of defining the test frequency, no change in the method of testing is being proposed. The Type A, B, and C tests will continue to be done at full pressure (P_a) or greater. Other programs are in place to ensure that proper maintenance and repairs are performed during the service life of the primary containment and systems and components penetrating the primary containment.

As a result, IP has concluded that the proposed change will not result in a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Vespasian Warner Public Library, 120 West Johnson Street, Clinton, Illinois 61727

Attorney for licensee: Leah Manning Stetener, Vice President, General Counsel, and Corporate Secretary, 500 South 27th Street, Decatur, Illinois 62525

NRC Project Director: Gail H. Marcus

Power Authority of the State of New York, Docket No. 50-333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of amendment request: January 25, 1996

Description of amendment request: The amendment proposes to extend instrumentation and miscellaneous surveillance test intervals (STI) to support 24-month operating cycles. Additionally, this application proposes: (1) to revise the Trip Level Settings for Emergency Bus Loss of Voltage and Degraded Voltage Instrumentation, (2) to revise the Reactor Protection System (RPS) Normal Supply Electrical Protection Assembly (EPA) Undervoltage Trip Setpoint, and (3) to make editorial revisions, clarification and Bases changes.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Operation of the FitzPatrick plant in accordance with the proposed Amendment would not involve a significant hazards consideration as defined in 10 CFR 50.92, since it would not:

1. involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed STI changes evaluated in Section IV.A do not involve any physical changes to the plant, do not alter the way these systems function, and will not degrade the performance of the plant safety systems. Proposed instrument setpoint changes ensure that plant safety limits are not exceeded due to instrument drift predicted for the longer calibration interval. The type of testing and the corrective actions required if the subject surveillances fail remains the same. The proposed changes do not adversely affect the reliability of these systems or affect the ability of the systems to meet their design objectives. A historical review of surveillance test results supports these conclusions.

The Trip Level Setpoint changes evaluated in Section IV.B ensure that the related systems perform as assumed in the transient and accident analysis by ensuring that plant safety limits are not exceeded due to instrument drift predicted for the longer calibration interval. The changes do not alter the system function, and will not degrade the performance of plant safety systems. The proposed Trip Level Setting changes do not adversely affect the reliability of these systems or adversely affect the ability of these systems to meet their design objectives.

The editorial, clarification and Bases changes evaluated in Section IV.C propose enhancements that clarify the Technical Specifications requirements and are editorial in nature. These changes do not alter any Technical Specification requirement, do not involve physical changes to the plant, or alter any operational setpoints. There are no safety implications in these proposed changes.

2. create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed STI changes evaluated in Section IV.A do not modify the design or operation of the plant, therefore, no new failure modes are introduced. Proposed instrument setpoint changes ensure that plant safety limits are not exceeded due to instrument drift resulting from the longer calibration interval. No changes are proposed to the type and method of testing performed, only to the length of the surveillance test interval. Past equipment performance and on-line testing indicate that longer test intervals will not degrade these systems. A historical review of surveillance test results supports these conclusions.

The Trip Level Setpoint changes evaluated in Section IV.B ensure that the related systems perform as assumed in the transient and accident analysis by ensuring that plant safety limits are not exceeded due to instrument drift predicted for the longer calibration interval. The changes do not alter the system function, introduce any new failure modes, and will not degrade the performance of plant safety systems. The proposed Trip Level Setting changes do not adversely affect the reliability of these systems or adversely affect the ability of these systems to meet their design objectives.

The editorial, clarification and Bases changes evaluated in Section IV.C propose enhancements that clarify the Technical Specifications requirements and are editorial in nature. These changes do not alter any Technical Specification requirement, do not involve physical changes to the plant, or alter any operational setpoints. There are no safety implications in these proposed changes.

3. involve a significant reduction in a margin of safety.

Although the proposed STI changes evaluated in Section IV.A will result in an increase in the interval between surveillance tests, the impact on system reliability is minimal. This is based on more frequent on-line testing and the redundant design of the evaluated systems. A review of past surveillance history has shown no evidence of failures which would significantly impact the reliability of these systems. Operation of the plant remains unchanged by these proposed STI extensions. The assumptions in the Plant Licensing Basis are not adversely impacted. Therefore, the proposed changes do not result in a significant reduction in the margin of safety.

The Trip Level Setpoint changes evaluated in Section IV.B ensure that the related systems perform as assumed in the transient and accident analysis by ensuring that plant safety limits are not exceeded due to instrument drift predicted for the longer calibration interval. The changes do not alter the system function, introduce any new failure modes, and will not degrade the performance of plant safety systems. The proposed Trip Level Setting changes do not adversely affect the reliability of these systems or adversely affect the ability of these systems to meet their design objectives.

The editorial, clarification and Bases changes evaluated in Section IV.C propose enhancements that clarify the Technical Specifications requirements and are editorial in nature. These changes do not alter any Technical Specification requirement, do not involve physical changes to the plant, or alter any operational setpoints. There are no safety implications in these proposed changes.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

Attorney for licensee: Mr. Charles M. Pratt, 1633 Broadway, New York, New York 10019.

NRC Project Director: Susan Frant Shankman, Acting

Power Authority of the State of New York, Docket No. 50-333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of amendment request: April 24, 1996

Description of amendment request: This amendment proposes to relocate Technical Specification (TS) 3.11.B/4.11.B ''Crescent Area Ventilation'' and associated Bases from the TS to an Authority controlled procedure.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Operation of the FitzPatrick plant in accordance with the proposed Amendment will not involve a significant hazards consideration as defined in 10 CFR 50.92, based on the following:

(1) These changes do not involve a significant increase in the probability or consequences of an accident previously evaluated because:

No modifications, no changes to operating procedure requirements, and no reduction in equipment reliability are being made as a result of these changes. Operating limitations will continue to be imposed, and required surveillance will continue to be performed in accordance with regulations, and written procedures and instructions that are auditable by the [Nuclear Regulatory Commission] NRC. Crescent Area Ventilation operability and testing requirements will continue to be an integral part of FitzPatrick plant operation.

Although future changes to the Crescent Area Ventilation system will no longer be controlled by 10 CFR 50.90, proposed changes will be evaluated under 10 CFR 50.59 and plant procedures. Programmatic controls will continue to assure that Crescent Area Ventilation system changes will not adversely affect [Emergency Core Cooling System] ECCS or [Reactor Core Isolation Cooling] RCIC system operability. As such, there is no significant increase in the probability or consequences of an accident previously evaluated.

(2) These changes do not create the possibility of a new or different type of accident previously evaluated because:

No modifications, no changes to operating procedure requirements, and no reduction in equipment reliability are being made as a result of these changes. Compliance with Crescent Area Ventilation system operability and surveillance requirements will be assured by maintaining them in an Authority controlled procedure. Changes to the Crescent Area Ventilation system will be subject to the requirements of 10 CFR 50.59. Therefore, the proposed changes do not introduce any failure mechanism of a different type than those previously evaluated since there are no changes being made to the facility and do not create the possibility of a new or different type of accident previously evaluated.

(3) The proposed amendment does not involve a reduction in a margin of safety because:

The Crescent Area Ventilation system supports Core Spray, [Low Pressure Coolant Injection] LPCI mode of [Residual Heat Removal] RHR, containment cooling mode of RHR, [High Pressure Coolant Injection] HPCI, and RCIC operability, and Crescent Area Ventilation system inoperability does affect these systems. As a result, the requirement for Crescent Area Ventilation to be operable for these systems to be considered operable is implicit in TS Sections 3.5.A, 3.5.B, 3.5.C, 3.5.E, and the definition of OPERABLE contained in TS Section 1.0.J. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

Attorney for licensee: Mr. Charles M. Pratt, 1633 Broadway, New York, New York 10019.

NRC Project Director: Susan Frant Shankman, Acting

Public Service Electric & Gas Company, Docket No. 50-311, Salem Nuclear Generating Station, Unit No. 2, Salem County, New Jersey

Date of amendment request: May 7, 1996

Description of amendment request: The proposed amendment involves a one-time change to Technical Specification (TS) 3/4.7.6, ''Control Room Emergency Air Conditioning System.'' The change would permit refueling of Salem, Unit 2, with the Control Room Emergency Air Conditioning System (CREACS) inoperable in Modes 5 and 6. The change will expire after the completion of the Control Room and CREACS upgrade, which is currently in progress, and the restart and entry into Mode 4 of Unit 2 from the current outage.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The CREACS is not an accident initiator. CREACS functions post-accident to provide cooling for Control Room equipment and habitability for operations personnel. Therefore, CREACS has no influence on the probability of any of the previously evaluated accidents or the other events evaluated as listed below.

Event

Fuel Handling Accident (Salem)

Waste Gas or Volume Control Tank Failures

Uncontrolled Boron Dilution

Loss of Offsite Power

Fuel Handling Accident (Hope Creek)

Liquid and Gaseous Waste Releases (Hope Creek)

Loss of Coolant Accident (LOCA) (Hope Creek)

Chemical Storage

Barge Collision

Control Room Internal and External Fire

Loss of Spent Fuel Pool Cooling

Loss of Decay Heat Removal

The Control Area Air Conditioning System (CAACS) and other measures will be available to maintain Control Room Envelope (CRE) ambient temperatures and habitability.

The proposed one-time change does not impact the consequences of an accident previously evaluated based on the following discussions.

The fuel has decayed to such low levels for more than six months that doses associated with the fuel handling accident are well within the limits of GDC [General Design Criteria] 19. There is insufficient activity remaining in either gaseous waste storage or liquid waste storage to force a Control Room evacuation. In the event of a Loss of Offsite Power (LOOP), uncontrolled boron dilution event, loss of spent fuel pool cooling or loss of decay heat removal, CREACS is not required in Modes 5 or 6 to mitigate the consequences of this event and CRE habitability will be maintained.

For a Hope Creek fuel handling accident, gaseous radwaste release of LOCA, dose to Salem Control Room personnel will not exceed GDC 19 limits. PSE&G [Public Service Electric & Gas] will maintain the CAACS [Control Area Air Conditioning System] outside air intakes either isolated or capable of being isolated in the event of a Hope Creek LOCA. The Hope Creek Event Classification Guide (ECG) requires notification of the Salem Control Room in the event of an emergency that has the potential to result in a radioactive release. The Salem Control Room will isolate the outside air intakes if isolation has not already been accomplished.

For the other events evaluated, the need for evacuation is not considered credible for any event with the exception of an internal or external fire. However, the possibility of evacuation of the CRE in the event of an internal or external fire would be no different whether or not CREACS is operating. In the event of an internal fire, CAACS will remain in operation to provide purging of the CRE. For the case of a possible external fire, the need for evacuation is not considered credible because of the short duration of the CREACS outage and improbability of the factors which are necessary to require an evacuation of the Control Room (i.e. wind direction, wind speed, amount of smoke). If an external fire is detected, operator action will be taken to isolate the CRE from outside air while CAACS remains available. In the unlikely event that the Control Room would become uninhabitable due to smoke in the atmosphere, evacuation procedures would be followed as in the case of the internal fire.

The one chemical storage type event which might impact the Control Room, rupture of an ammonium hydroxide tanker, is precluded by administrative controls such that no ammonium hydroxide tanker deliveries will be allowed during the system upgrade period.

The CAACS will maintain the current design function and TS Bases requirements of the CREACS that the ambient air temperature does not exceed the allowable temperature for continuous duty rating for equipment and instrumentation cooled by the system for the combined CRE. The CAACS will be maintained functional while modification to the CREACS is ongoing to provide cooling during normal operation and under postulated accident conditions. Should the temperature in the CRE exceed allowable levels (85 Degrees F), administrative controls will be in place to require restoration of the temperature to within acceptable levels using CAACS, and prevent any Core Alteration activities or positive reactivity changes until the temperature is restored to acceptable levels.

Therefore, the proposed one-time TS change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The CREACS is not an accident initiator. CREACS functions post-accident to provide cooling for Control Room equipment and habitability for operations personnel. Therefore, CREACS inoperability during Modes 5 and 6 will not result in the creation of a new or different kind of accident from any accident previously evaluated. All pertinent accidents have been assessed and no other scenarios dealing with fuel movement, or the need for an operable CREACS in Mode 5 or 6, have been deemed credible.

Therefore, the proposed one-time change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change does not involve a significant reduction in a margin of safety.

The proposed one-time change does not significantly reduce the margin of safety as defined in the Bases for the TS because (1) there is no credible event as analyzed in Salem UFSAR [updated final safety analysis report] Chapter 15 which can cause an unacceptable environment in the CRE since the fuel has been decaying for at least six months, (2) fuel movement inside the Fuel Handling Building (FHB) is restricted in accordance with plant TS unless FHB ventilation is operable, (3) dose to Salem control room personnel from a potential Hope Creek fuel handling accident, gaseous radwaste release or Loss of Coolant Accident will not exceed GDC 19 limits (4) the one event which might impact the Control Room, rupture of an ammonium hydroxide tanker, is precluded by administrative controls such that no ammonium hydroxide tanker deliveries will be allowed during the CREACS upgrade period, and (5) in the unlikely event that Control Room evacuation is required, there is no impact on operator ability to mitigate the consequences of an accident in the current plant configuration.

Therefore, the proposed one-time TS change does not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Salem Free Public library, 112 West Broadway, Salem, New Jersey 08079

Attorney for licensee: Mark J. Wetterhahn, Esquire, Winston and Strawn, 1400 L Street, NW, Washington, DC 20005-3502

NRC Project Director: John F. Stolz

Southern Nuclear Operating Company, Inc., Docket No. 50-364, Joseph M. Farley Nuclear Plant, Unit 2, Houston County, Alabama

Date of amendment request: March 29, 1996

Description of amendment request: The proposed amendment would revise Technical Specification 3/4.4.6 ''Steam Generators'' and its associated Bases. Specifically, the steam generator repair limit would be modified to clarify that the appropriate method for determining serviceability for tubes with outside diameter stress corrosion cracking at the tube support plate is by a methodology that more reliably assesses structural integrity. This amendment request is in accordance with NRC's Generic Letter 95-05, ''Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking.''

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Operation of Farley units in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Testing of model boiler specimens for free standing tubes at room temperature conditions shows burst pressures as high as approximately 5000 psi for indications of outer diameter stress corrosion cracking with voltage measurements as high as 26.5 volts. Burst testing performed on pulled tubes, including tubes pulled from Farley Unit 2, with up to 7.5 volt indications show burst pressures in excess of 5300 psi at room temperature. As stated earlier, tube burst criteria are inherently satisfied during normal operating conditions by the presence of the tube support plate. Furthermore, correcting for the effects of temperature on material properties and minimum strength levels (as the burst testing was done at room temperature), tube burst capability significantly exceeds the R.G. [Regulatory Guide] 1.121 criterion requiring the maintenance of a margin of 1.43 times the steam line break pressure differential on tube burst if through-wall cracks are present without regard to the presence of the tube support plate. Considering the existing data base, this criterion is satisfied with bobbin coil indications with signal amplitudes over twice the 2.0 volt voltage-based repair criteria, regardless of the indicated depth measurement. This structural limit is based on a lower 95% confidence level limit of the data at operating temperatures. The 2.0 volt criterion provides a conservative margin of safety to the structural limit considering expected growth rates of outside diameter stress corrosion cracking at Farley. Alternate crack morphologies can correspond to a voltage so that a unique crack length is not defined by a burst pressure to voltage correlation. However, relative to expected leakage during normal operating conditions, no field leakage has been reported from tubes with indications with a voltage level of under 7.7 volts for a 3/4 inch tube with a 10 volt correlation to 7/8 inch tubing (as compared to the 2.0 volt proposed voltage-based tube repair limit). Thus, the proposed amendment does not involve a significant increase in the probability or consequences of an accident.

Relative to the expected leakage during accident condition loadings, the accidents that are affected by primary-to-secondary leakage and steam release to the environment are Loss of External Electrical Load and/or Turbine Trip, Loss of All AC Power to Station Auxiliaries, Major Secondary System Pipe Failure, Steam Generator Tube Rupture, Reactor Coolant Pump Locked Rotor, and Rupture of a Control Rod Drive Mechanism Housing. Of these, the Major Secondary System Pipe Failure is the most limiting for Farley in considering the potential for off-site doses. The offsite dose analyses for the other events which model primary-to secondary leakage and steam releases from the secondary side to the environment assume that the secondary side remains intact. The steam generator tubes are not subjected to a sustained increase in differential pressure, as is the case following a steam line break event. This increase in differential pressure is responsible for the postulated increase in leakage and associated offsite doses following a steam line break event. In addition, the steam line break event results in a bypass of containment for steam generator leakage. Upon implementation of the voltage-based repair criteria, it must be verified that the expected distributions of cracking indications at the tube support plate intersections are such that primary-to-secondary leakage would result in site boundary dose within the current licensing basis. Data indicate that a threshold voltage of 2.8 volts could result in through-wall cracks long enough to leak at steam line break conditions. Application of the proposed repair criteria requires that the current distribution of a number of indications versus voltage be obtained during the refueling outages. The current voltage is then combined with the rate of change in voltage measurement and a voltage measurement uncertainty to establish an end of cycle voltage distribution and, thus, leak rate during steam line break pressure differential. The leak rate during a steam line break is further increased by a factor related to the probability of detection of the flaws. If it is found that the potential steam line break leakage for degraded intersections planned to be left in service coupled with the reduced allowable specific activity levels result in radiological consequences outside the current licensing basis, then additional tubes will be plugged or repaired to reduce steam line break leakage potential to within the acceptance limit. Thus, the consequences of the most limiting design basis accident are constrained to present licensing basis limits.

2) The proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Implementation of the proposed voltage-based tube support plate elevation steam generator tube repair criteria does not introduce any significant changes to the plant design basis. Use of the criteria does not provide a mechanism that could result in an accident outside of the region of the tube support plate elevations. Neither a single or multiple tube rupture event would be expected in a steam generator in which the repair criteria have been applied during all plant conditions. The bobbin probe signal amplitude repair criteria are established such that operational leakage or excessive leakage during a postulated steam line break condition is not anticipated. Southern Nuclear has previously implemented a maximum leakage limit of 150 gpd per steam generator. The R.G. 1.121 criterion for establishing operational leakage limits that require plant shutdown are based upon leak-before-break considerations to detect a free span crack before potential tube rupture. The 150 gpd limit provides for leakage detection and plant shutdown in the event of the occurrence of an unexpected single crack resulting in leakage that is associated with the longest permissible crack length. R.G. 1.121 acceptance criteria for establishing operating leakage limits are based on leak-before-break considerations such that plant shutdown is initiated if the leakage associated with the longest permissible crack is exceeded. The longest permissible crack is the length that provides a factor of safety of 1.43 against bursting at steam line break pressure differential. A voltage amplitude of approximately 9 volts for typical outside diameter stress corrosion cracking corresponds to meeting this tube burst requirement at the 95% prediction interval on the burst correlation. Alternate crack morphologies can correspond to a voltage so that a unique crack length is not defined by the burst pressure versus voltage correlation. Consequently, a typical burst pressure versus through-wall crack length correlation is used below to define the ‘‘longest permissible crack’’ for evaluating operating leakage limits.

The single through-wall crack lengths that result in tube burst at 1.43 times steam line break pressure differential and steam line break conditions are about 0.54 inch and 0.84 inch, respectively. Normal leakage for these crack lengths would range from about 0.4 gallons per minute to 4.5 gallons per minute, respectively, while lower 95% confidence level leak rates would range from about 0.06 gallons per minute to 0.6 gallons per minute, respectively.

An operating leak rate of 150 gpd per steam generator has been implemented. This leakage limit provides for detection of 0.4 inch long cracks at nominal leak rates and 0.6 inch long cracks at the lower 95% confidence level leak rates. Thus, the 150 gpd limit provides for plant shutdown prior to reaching critical crack lengths for steam line break conditions at leak rates less than a lower 95% confidence level and for three times normal operating pressure differential at less than nominal leak rates.

Considering the above, the implementation of voltage-based plugging criteria will not create the possibility of a new or different kind of accident from any previously evaluated.

3) The proposed license amendment does not involve a significant reduction in margin of safety.

The use of the voltage-based tube support plate elevation repair criteria is demonstrated to maintain steam generator tube integrity commensurate with the requirements of Generic Letter 95-05 and R.G. 1.121. R.G. 1.121 describes a method acceptable to the NRC staff for meeting GDC [Generic Design Criteria] 2, 14, 15, 31, and 32 by reducing the probability of the consequences of steam generator tube rupture. This is accomplished by determining the limiting conditions of degradation of steam generator tubing, as established by inservice inspection, for which tubes with unacceptable cracking should be removed from service. Upon implementation of the criteria, even under the worst case conditions, the occurrence of outside diameter stress corrosion cracking at the tube support plate elevations is not expected to lead to a steam generator tube rupture event during normal or faulted plant conditions. The most limiting effect would be a possible increase in leakage during a steam line break event. Excessive leakage during a steam line break event, however, is precluded by verifying that, once the criteria are applied, the expected end of cycle distribution of crack indications at the tube support plate elevations would result in minimal, and acceptable primary to secondary leakage during the event and, hence, help to demonstrate radiological conditions are less than an appropriate fraction of the 10 CFR [Part] 100 guideline.

The margin to burst for the tubes using the voltage-based repair criteria is comparable to that currently provided by existing Technical Specifications.

In addressing the combined effects of LOCA [loss-of-coolant accident] + SSE [safe-shutdown earthquake] on the steam generator component (as required by GDC 2), it has been determined that tube collapse may occur in the steam generators at some plants. This is the case as the tube support plates may become deformed as a result of lateral loads at the wedge supports at the periphery of the plate due to either the LOCA rarefaction wave and/or SSE loadings. Then, the resulting pressure differential on the deformed tubes may cause some of the tubes to collapse.

There are two issues associated with steam generator tube collapse. First, the collapse of steam generator tubing reduces the RCS [reactor coolant system] flow area through the tubes. The reduction in flow area increases the resistance to flow of steam from the core during a LOCA which, in turn, may potentially increase Peak Clad Temperature (PCT). Second, there is a potential the partial through-wall cracks in tubes could progress to through-wall cracks during tube deformation or collapse or that short through-wall indications would leak at significantly higher leak rates than included in the leak rate assessments.

Consequently, a detailed leak-before-break analysis was performed and it was concluded that the leak-before-break methodology (as permitted by GDC 4) is applicable to the Farley reactor coolant system primary loops and, thus, the probability of breaks in the primary loop piping is sufficiently low that they need not be considered in the structural design basis of the plant. Excluding breaks in the RCS primary loops, the LOCA loads from the large branch line breaks were analyzed at Farley and were found to be of insufficient magnitude to result in steam generator tube collapse or significant deformation.

Regardless of whether or not leak-before-break is applied to the primary loop piping at Farley, any flow area reduction is expected to be minimal (much less than 1%) and PCT margin is available to account for this potential effect. Based on analyses' results, no tubes near wedge locations are expected to collapse or deform to the degree that secondary to primary in-leakage would be increased over current expected levels. For all other steam generator tubes, the possibility of secondary-to-primary leakage in the event of a LOCA + SSE event is not significant. In actuality, the amount of secondary-to-primary leakage in the event of a LOCA + SSE is expected to be less than that originally allowed, i.e., 500 gpd per steam generator. Furthermore, secondary-to-primary in-leakage would be less than primary-to-secondary leakage for the same pressure differential since the cracks would tend to tighten under a secondary-to-primary pressure differential. Also, the presence of the tube support plate is expected to reduce the amount of in-leakage.

Addressing the R.G. 1.83 considerations, implementation of the tube repair criteria is supplemented by 100% inspection requirements at the tube support plate elevations having outside diameter stress corrosion cracking indications, reduced operating leakage limits, eddy current inspection guidelines to provide consistency in voltage normalization, and rotating probe inspection requirements for the larger indications left in service to characterize the principle degradation mechanism as outside diameter stress corrosion cracking.

As noted previously, implementation of the tube support plate elevation repair criteria will decrease the number of tubes that must be taken out of service with tube plugs or repaired. The installation of steam generator tube plugs or tube sleeves would reduce the RCS flow margin, thus implementation of the voltage-based repair criteria will maintain the margin of flow that would otherwise be reduced through increased tube plugging or sleeving.

Considering the above, it is concluded that the proposed change does not result in a significant reduction in margin with respect to plant safety as defined in the Final Safety Analysis Report or any bases of the plant Technical Specifications.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Houston-Love Memorial Library, 212 W. Burdeshaw Street, Post Office Box 1369, Dothan, Alabama 36302

Attorney for licensee: M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201

NRC Project Director: Herbert N. Berkow

Southern Nuclear Operating Company, Inc., Docket No. 50-364, Joseph M. Farley Nuclear Plant, Unit 2, Houston County, Alabama

Date of amendment request: April 22, 1996

Description of amendment request: The proposed amendment would implement a new F* criterion based on maintaining existing safety margins for steam generator tube structural integrity concurrent with allowance for NDE (nondestructive examination) eddy current uncertainty.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed change retains the existing margin in the F* distance used to meet regulatory guidance of draft Regulatory Guide 1.121 and only changes the amount of assumed NDE eddy current uncertainty based on the type of eddy current technology utilized in the inspection. Therefore, there is no significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed change does not create the possibility of a new or different kind of accident from any previously evaluated. WCAP 11306, Revision 2, ''Tubesheet Region Plugging Criterion for the Alabama Power Company Farley Nuclear Station Unit 2 Steam Generators,'' provides adequate basis for the F* distance proposed of 1.54 plus allowance for eddy current uncertainty measurement. Since the value of 1.54 inches was used in the analysis no new or different kind of accident from any accident previously evaluated will be created.

3. The proposed change does not involve a significant reduction in a margin safety. Since the value of 1.54 inches already is used in the steam generator tube pull out analysis, there is no significant change to a margin safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Houston-Love Memorial Library, 212 W. Burdeshaw Street, Post Office Box 1369, Dothan, Alabama 36302

Attorney for licensee: M. Stanford Blanton, Esq., Balch and Bingham, Post Office Box 306, 1710 Sixth Avenue North, Birmingham, Alabama 35201

NRC Project Director: Herbert N. Berkow

Union Electric Company, Docket No. 50-483, Callaway Plant, Unit 1, Callaway County, MissouriDate of application request: February 23, 1996, as supplemented by letter dated April 24, 1996.

Description of amendment request: The amendment would add a footnote in the license for Callaway Plant, Unit No. 1 to indicate that Union Electric Company has entered into a merger agreement with CIPSCO Incorporated which provides for Union Electric Company to become a wholly-owned operating company of Ameren Corporation, a registered public utility holding company under the Public Utility Holding Company Act of 1935, as amended. After the merger, Union Electric Company would continue to own and operate the Callaway Plant as an operating company subsidiary of Ameren Corporation.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change does not affect accident initiators or assumptions. The radiological consequences of any accident previously evaluated remain unchanged. The change is an administrative change to reflect Union Electric's status as an operating company subsidiary of Ameren.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change does not reduce the margin of safety assumed in any accident analysis or affect any safety limits. The change is administrative and reflects Union Electric's status as an operating company subsidiary of Ameren.

3. The proposed change does not involve a significant reduction in a margin of safety.

The proposed change does not reduce the margin of safety assumed in any accident analysis or affect any safety limits. The change is administrative and reflects Union Electric's status as an operating company subsidiary of Ameren.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Callaway County Public Library, 710 Court Street, Fulton, Missouri 65251

Attorney for licensee: Gerald Charnoff, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037

NRC Project Director: William H. Bateman

Wisconsin Public Service Corporation, Docket No. 50-305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin

Date of amendment request: April 30, 1996

Description of amendment request: The proposed amendment would revise Kewaunee Nuclear Power Plant (KNPP) Technical Specification (TS) 3.1.b.1, its associated bases, and Figure TS 3.1-4 by extending the low temperature overpressure protection (LTOP) requirements through the end of operating cycle 33 or 33.41 effective full power years. The only technical change being proposed is the substitution of end of life fluence for the end of operating cycle 21 fluence.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

The proposed change was reviewed in accordance with the provisions of 10 CFR 50.92 to show no significant hazards exist. The proposed change will not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

The LTOP setpoint and revised P/T [pressure/temperature] limits reflected in proposed Figure TS 3.1-4 ensure that the Appendix G pressure/temperature limits are not exceeded, and therefore, help ensure that RCS integrity is maintained. The changes do not modify the reactor coolant system pressure boundary, nor make any physical changes to the facility design, material, construction standards, or setpoints. The LTOP valve setpoint remains set at 500 psi. The LTOP enabling temperature based on Figure TS 3.1-2 is 338°F and is more conservative than a value of 303° Figure TS 3.1-4. The LTOP enabling temperature based on Figure TS 3.1-2 remains unchanged by this PA [proposed amendment]. The probability of a LTOP event occurring is independent of the pressure-temperature limits for the RCS pressure boundary. Therefore, the probability of a LTOP event occurring remains unchanged.

The calculation of pressure temperature limits in accordance with approved regulatory methods provides assurance that reactor pressure vessel fracture toughness requirements are met and the integrity of the RCS [reactor coolant system] pressure boundary is maintained. Similar methodology was used in calculations to support approved amendment 120 to the Kewaunee Technical Specifications dated April 26, 1995. The material property basis, including chemistry factor and initial reference temperature for the unirradiated material (RT_{NDT}), used for this PA is the same as that used in the current TS. The only technical change being made in this PA is the use of end of life fluence.

The use of predicted fluence values through the end of operating cycle 33 is appropriately considered within the calculations in accordance with standard industry methodology previously docketed under WCAP 13227 and WCAP 14279. The neutron exposure projections utilized for calculation of the reference temperature were multiplied by a factor of 1.11 to adjust for biases observed between cycle specific calculations and the results of neutron dosimetry for the four surveillance capsules removed from the KNPP reactor. The factor of 1.11 was derived by taking the average of the measured to calculation (M/C) flux ratios obtained from the dosimetry results of capsules V, R, P, and S removed from the KNPP reactor vessel. The resulting effect of using predicted fluence values through the end of cycle 33 instead of cycle 21 is to require the plant to evaluate LTOP transients to more limiting requirements. The proposed PT limits are shifted to a lower pressure and higher temperature, which is more conservative.

The changes do not adversely affect the integrity of the RCS such that its function in the control of radiological consequences is affected. In addition, the changes do not affect any fission barrier. The changes do not degrade or prevent the response of the LTOP relief valve or other safety related system to accidents described in Chapter 14 of the USAR. In addition, the changes do not alter any assumption previously made in the radiological consequences evaluations nor affect the mitigation of the radiological consequences of an accident described in the USAR. Therefore, the consequences of an accident previously evaluated in the USAR will not be increased.

Thus, the operation of KNPP Unit 1 in accordance with the PA does not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Create the possibility of a new or different type of accident from an accident previously evaluated.

The Appendix G pressure temperature limitations were prepared using methods derived from the ASME Boiler and Pressure Vessel Code and the criteria set forth in NRC Regulatory Standard Review Plan 5.3.2. The changes do not cause the initiation of any accident nor create any new credible limiting failure for safety-related systems and components. The changes do not result in any event previously deemed incredible being made credible. As such, it does not create the possibility of an accident different than any evaluated in the USAR.

The changes do not have any effect on the ability of the safety-related systems to perform their intended safety functions. The changes do not create failure modes that could adversely impact safety-related equipment. Therefore, it will not create the possibility of a malfunction of equipment important to safety different than previously evaluated in the USAR. Thus, the PA does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The use of Paragraph (c)(2)(ii)(A) of 10 CFR 50.61, initial reference temperature of -50°F, and the fluence values through EOC [end of cycle] 33 does not modify the reactor coolant system pressure boundary, nor make any physical changes to the LTOP setpoint or system design. Proposed Figure TS 3.1-4 was prepared in accordance with regulatory requirements and requires evaluation of LTOP events to more limiting requirements of neutron exposure projections of 33.41 EFPY instead of 18.40 EFPY.

Therefore, the PA does not create the possibility of a new or different type of accident from any accident previously evaluated.

3. Involve a significant reduction in the margin of safety.

The Appendix G pressure temperature limitations were prepared using methods derived from the ASME Boiler and Pressure Vessel Code and the criteria set forth in NRC Regulatory Standard Review Plan 5.3.2. These documents along with the calculational limitations specified in 10 CFR 50.61 are an acceptable method for implementing the requirements of 10 CFR 50 Appendices G and H. Inherent conservatism in the P/T limits resulting from these documents include:

a. An assumed defect in the reactor vessel wall with a depth equal to 1/4 of the thickness of the vessel wall (1/4T) and a length equal to 1-1/2 times the thickness of the vessel wall.

b. Assumed reference flaw oriented in both longitudinal and circumferential directions and limiting material property. At KNPP, the only weld in the core region is oriented in the circumferential direction.

c. A factor of safety of 2 is applied to the membrane stress intensity factor.

d. The limiting toughness is based upon a reference value (K_{IR}) which is a lower bound on the dynamic crack initiation or arrest toughness.

e. A 2-sigma margin term is applied in determining the adjusted reference temperature (ART) that is used to calculate the limiting toughness.

Similar methodology was used in calculations to support approved amendment 120 dated April 26, 1995. Beyond the conservatism described above, WPSC

[Wisconsin Public Service Corporation] has incorporated the following additional margin in preparing this PA:

a. The neutron exposure projections were multiplied by a factor of 1.11 to adjust for biases observed between cycle specific calculations and the results of neutron dosimetry for the four surveillance capsules removed from the KNPP reactor. The factor of 1.11 was derived by taking the average of the measured to calculation (M/C) flux ratios obtained from the dosimetry results of capsules V, R, P, and S removed from the KNPP reactor vessel.

b. The calculated material-specific chemistry factor value is 191.27 and is based on KNPP surveillance capsule data from capsules V, R, and P. Utilization of KNPP's most recent surveillance capsule data from capsule S results in chemistry factor value of 190.6. Consistent with calculation C10689, Revision 1 the value used for chemistry factor in this PA remains 191.27, which is conservative.

c. The LTOP enabling temperature based on Figure TS 3.1-2 is 338°F and is more conservative than a value of 303°F which is supported by proposed Figure TS 3.1-4. The LTOP enabling temperature based on Figure TS 3.1-2 remains unchanged by this PA.

d. The reactor coolant pump starting restrictions of TS 3.1.a.1.c remain in place.

An alternative methodology to the safety margins required by Appendix G to 10 CFR Part 50 has been developed by the ASME Working Group on Operating Plant Criteria. This methodology is contained in ASME Code Case N-514. The Code Case N-514 provides criteria to determine pressure limits during LTOP events that avoid certain unnecessary operational restrictions, provide adequate margins against failure of the reactor pressure vessel, and reduce the potential for unnecessary activation of the relief valve used for LTOP. Specifically, the ASME Code Case N-514 allows determination of the setpoint for LTOP events such that the maximum pressure in the vessel would not exceed 110% of the P/T limits of the existing ASME Appendix G; and redefines the enabling temperature as a coolant temperature less than 200°F or a reactor vessel metal temperature less than RT_{NDT} + 50°F greater. Code Case N-514, ''Low Temperature Overpressure Protection,'' has been approved by the ASME Code Committee but not yet approved for use in Regulatory Guide 1.147. The content of this code case has been incorporated into Appendix G of Section XI of the ASME Code and published in the 1993 Addenda to Section XI. It is expected that when the NRC revises 10 CFR 50.55a, it will endorse the 1993 Addenda and Appendix G of Section XI into the regulations. As stated above, this PA utilizes Appendix G limits and an enabling temperature corresponding to a reactor vessel metal temperature less than RT_{NDT} + 90°F, which is more conservative than the alternative methodology contained in Code Case N-514.

The revised calculations meet the NRC acceptance criteria for the LTOP setpoint and system design as described in NRC Safety Evaluation Report (SER) dated September 6, 1995 which concluded that ''the spectrum of postulated pressure transients would be mitigated...such that the temperature pressure limits of Appendix G to 10 CFR 50 are maintained.''

Utilization of methodology set forth in the ASME Boiler and Pressure Vessel Code, NRC Regulatory Standard Review Plan 5.3.2, 10 CFR 50.61, and 10 CFR 50 Appendices G and H with the above additional margins ensures that proper limits and safety factors are maintained. Thus, the PA does not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: University of Wisconsin, Cofrin Library, 2420 Nicolet Drive, Green Bay, Wisconsin 54311-7001

Attorney for licensee: Bradley D. Jackson, Esq., Foley and Lardner, P. O. Box 1497, Madison, Wisconsin 53701-1497

NRC Project Director: Gail H. Marcus

Wisconsin Public Service Corporation, Docket No. 50-305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin

Date of amendment request: May 1, 1996

Description of amendment request: The proposed amendment would revise Kewaunee Nuclear Power Plant (KNPP) Technical Specification (TS) 4.2.b, ''Steam Generator Tubes,'' its associated bases, and Figure TS 4.2-1 by redefining the pressure boundary for Westinghouse mechanical hybrid expansion joint (HEJ) steam generator (SG) tube sleeves. The proposed amendment supersedes in its entirety a previously submitted proposed amendment dated October 6, 1995, which was published in the Federal Register on November 8, 1995 (60 FR 56372).

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

This proposed change was reviewed in accordance with the provisions of 10 CFR 50.92 to show no significant hazards exist.

1. Operation of the KNPP in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Mechanical testing shows inherent structural integrity of the HEJ [hybrid expansion joint] upper joint such that the tube rupture capability recommendations of RG [Regulatory Guide] 1.121 are met, even for instances of 100-percent throughwall, 360 degree degradation in the HRLT [hardroll lower transition] region. Structural test results are documented in WCAPs-14157, -14157 Addendum 1, -14446 and -14641. Based on this test data, the structural recommendations of RG 1.121 are satisfied when there is a difference of at least 0.003 inch, between the maximum hardroll diameter of the sleeve, and the diameter at the elevation of the PTI [parent tube indication] center line; i.e. there is an interference lip of 0.003 inch or more. The proposed pressure boundary will allow PTIs located such that there is a minimum diameter change of 0.003 inch (not including an allowance for measurement uncertainty) between the maximum point of the sleeve hardroll, and the diameter at the elevation of the PTI peak amplitude to remain in service. Based on the high degree of structural integrity of the HEJ upper joint, it can be concluded that application of the revised pressure boundary criteria will not result in an increased probability of an accident previously evaluated.

Each sleeved tube with a PTI located in the HRLT such that there is a change in diameter of 0.003 inch to 0.013 inch, will be assigned a conservatively bounding primary-to-secondary SLB [steam line break] leakage value of 0.025 gpm per indication. Indications located such that there is a change in diameter of greater than 0.013 inch will not contribute to the SLB leakage. The total number of indications remaining in service will be limited such that the primary-to-secondary leakage during a postulated SLB will not exceed a small fraction of the 10 CFR Part 100 guidelines. For KNPP this has been calculated to be 34.0 gpm for the faulted loop. Therefore, it can be concluded that application of the revised pressure boundary criteria will not increase the consequences of an accident previously evaluated.

2. The proposed license amendment request does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Implementation of the revised pressure boundary will not introduce a change to the design basis or operation of the plant. Mechanical testing of degraded sleeve joints supports the conclusions that the joint retains structural integrity (tube burst) capability consistent with RG 1.121, and leakage integrity with regards to a small fraction of the 10 CFR Part 100 guidelines. As with the initial installation of the sleeves, implementation of the relocated pressure boundary does not interact with other portions of the reactor coolant system. Any hypothetical accident as a result of potential PTIs is bounded by the existing tube rupture accident analysis. Neither the sleeve design nor implementation of the redefined pressure boundary affects any other component or location of the tube outside of the immediate area repaired. Therefore application of the revised pressure boundary criteria will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed license amendment does not involve a significant reduction in the margin of safety.

The safety factors used in establishment of the HEJ sleeved tube pressure boundary are consistent with the safety factors in the ASME Boiler and Pressure Vessel Code used in SG [steam generator] design. Based on the sleeve-to-tube geometry, it is unrealistic to consider that application of the revised pressure boundary could result in single tube leak rates exceeding the normal makeup capacity during normal operating conditions. The pressure boundary developed in WCAPs-14446 and -14641 have been developed using the methodology of RG 1.121. The performance characteristics of the postulated degraded parent tubes of HEJ sleeve/tube joints have been verified by testing to retain structural integrity and preclude significant leakage during normal and postulated accident conditions. Testing indicates that postulated circumferentially separated tubes which the pressure boundary [addresses] would not experience axial displacement during either normal operation or SLB conditions. The existing offsite dose evaluation performed for KNPP in support of the voltage based repair criteria for axial ODSCC [outside diameter stress corrosion cracking] at TSP [tube support plate] intersections established a faulted loop primary to secondary leak rate of 34.0 gpm. Following implementation of the criteria, postulated leakage from all sources must not exceed 34.0 gpm in the faulted loop. Maintenance of this limit will ensure that offsite doses would not exceed the currently accepted limit of a small fraction of the 10 CFR Part 100 guidelines. The pressure boundary definition uses a conservatively established ‘‘per indication’’ leak rate for estimation of SLB leakage. This leak rate is applied to all indications left in service within the HRLT, regardless of indications length and throughwall extent. Application of the revised pressure boundary criteria will not result in a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee’s analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: University of Wisconsin, Cofrin Library, 2420 Nicolet Drive, Green Bay, Wisconsin 54311-7001.

Attorney for licensee: Bradley D. Jackson, Esq., Foley and Lardner, P. O. Box 1497, Madison, Wisconsin 53701-1497

NRC Project Director: Gail H. Marcus

Wolf Creek Nuclear Operating Corporation, Docket No. 50-482, Wolf Creek Generating Station, Coffey County, Kansas

Date of amendment request: July 29, 1994, as superseded by letter dated September 15, 1995, and supplements dated March 8, 1996, and April 18, 1996

Description of amendment request: The proposed amendment revises TS 3/4.8.1 and its associated Bases to improve overall emergency diesel generator reliability and availability.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

These proposed changes do not involve a change in the operational limits or physical design of the emergency power system. Emergency diesel generator operability and reliability will continue to be assured while minimizing the number of required emergency diesel generator starts. Also, emergency diesel generator reliability will be enhanced by minimizing severe test conditions which can lead to premature failures.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

These proposed changes do not involve a change in the operational limits or physical design of the emergency power system. The performance capability of the emergency diesel generator will not be affected. Emergency diesel generator reliability and availability will be improved by the implementation of the proposed changes. There is no actual impact on any accident analysis.

3. The proposed change does not involve a significant reduction in a margin of safety.

These proposed change do not involve a change in the operational limits or physical design of the emergency power system. The performance capability of the emergency diesel generator will not be affected. Emergency diesel generator reliability and availability will be improved by the implementation of the proposed changes. No margin of safety is reduced.

The NRC staff has reviewed the licensee’s analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room locations: Emporia State University, William Allen White Library, 1200 Commercial Street, Emporia, Kansas 66801 and Washburn University School of Law Library, Topeka, Kansas 66621

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037

NRC Project Director: William H. Bateman

Wolf Creek Nuclear Operating Corporation, Docket No. 50-482, Wolf Creek Generating Station, Coffey County, Kansas

Date of amendment request: May 1, 1996

Description of amendment request: This license amendment request proposes to revise Section 6.0 of the technical specifications to reflect position title changes within the Wolf Creek Nuclear Operating Corporation (WCNOC) organization.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change does not involve a significant increase in the probability of consequences of an accident previously evaluated. These changes involve administrative changes to the WCNOC organization and to the position qualification of plant personnel.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. This change is administrative in nature and does not involve a change to the installed plant systems or the overall operating philosophy of Wolf Creek Generating Station.

3. The proposed change does not involve a significant reduction in a margin of safety.

The proposed change does not involve a significant reduction in a margin of safety. This change does not involve any changes in overall organizational commitments. A position title change alone does not reduce the margin of safety.

The NRC staff has reviewed the licensee’s analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room locations: Emporia State University, William Allen White Library, 1200 Commercial Street, Emporia, Kansas 66801 and Washburn University School of Law Library, Topeka, Kansas 66621

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, N.W., Washington, D.C. 20037

NRC Project Director: William H. Bateman

Previously Published Notices Of Consideration Of Issuance Of Amendments To Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, And Opportunity For A Hearing

The following notices were previously published as separate individual notices. The notice content was the same as above. They were published as individual notices either because time did not allow the Commission to wait for this biweekly notice or because the action involved exigent circumstances. They are repeated here because the biweekly notice lists all amendments issued or proposed to be issued involving no significant hazards consideration.

For details, see the individual notice in the Federal Register on the day and page cited. This notice does not extend the notice period of the original notice.

Philadelphia Electric Company, Docket Nos. 50-352 and 50-353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of amendment request: April 25, 1996

Brief description of amendment request: The amendment relocates the technical specification (TS) Traversing In-Core Probe System Limiting Condition for Operation 3/4.3.7.7 and its Bases 3/4.3.7.7 to the Technical Requirements Manual, and modifies Note (f) of TS Table 4.3.1.1-1.

Date of publication of individual notice in Federal Register: May 8, 1996 (61 FR 20840)

Expiration date of individual notice: June 7, 1996

Local Public Document Room location: Pottstown Public Library, 500 High Street, Pottstown, Pennsylvania 19464.

The Cleveland Electric Illuminating Company, Centerior Service Company, Duquesne Light Company, Ohio Edison Company, OES Nuclear, Inc., Pennsylvania Power Company, Toledo Edison Company, Docket No. 50-440, Perry Nuclear Power Plant, Unit No. 1, Lake County, Ohio

Date of application for amendment: April 26, 1996

Brief description of amendment request: The proposed amendment would correct minor technical and administrative errors in the Improved Technical Specifications prior to its implementation.

Date of individual notice in Federal Register: May 9, 1996 (61 FR 21213)

Expiration date of individual notice: June 10, 1996

Local Public Document Room location: Perry Public Library, 3753 Main Street, Perry, Ohio

Notice Of Issuance Of Amendments To Facility Operating Licenses

During the period since publication of the last biweekly notice, the Commission has issued the following amendments. The Commission has determined for each of these amendments that the application complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for A Hearing in connection with these actions was published in the Federal Register as indicated.

Unless otherwise indicated, the Commission has determined that these amendments satisfy the criteria for categorical exclusion in accordance with 10 CFR 51.22. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for these amendments. If the Commission has prepared an environmental assessment under the special circumstances provision in 10 CFR 51.12(b) and has made a determination based on that assessment, it is so indicated.

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document rooms for the particular facilities involved.

Arizona Public Service Company, et al., Docket Nos. STN 50-528, STN 50-529, and STN 50-530, Palo Verde Nuclear Generating Station, Units 1, 2, and 3, Maricopa County, Arizona

Date of application for amendments: February 1, 1996

Brief description of amendments: These amendments revised (1) Technical Specifications (TS) 3/4.1.1.1, 6.9.1.9, and 6.9.1.10 to relocate the shutdown margin (reactor trip breakers open) to the Core Operating Limits Report; (2) TS 3/4.3.2 (Tables 3.3-3 and 3.3-4) to specify an additional restriction for the allowed low-pressurizer-pressure trip setpoint when reducing reactor coolant (RCS) system pressure in Mode 3; (3) TS Section 2.2.1 (Table 2.2-1) to make it consistent with the footnote in TS Tables 3.3-3 and 3.3-4; and (4) TS Sections 3/4.5.2 and 3/4.5.3 to require two emergency core cooling system subsystems to be operable in Mode 3 whenever the RCS cold-leg temperature is equal to or above 485°F. The Table of Contents and the Bases are also revised to reflect these changes.

Date of issuance: April 30, 1996

Effective date: April 30, 1996, to be implemented within 45 days of issuance

Amendment Nos.: Unit 1 - 106; Unit 2 - 98; Unit 3 - 78

Facility Operating License Nos. NPF-41, NPF-51, and NPF-74: The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: March 27, 1996 (61 FR 13522) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 30, 1996.No significant hazards consideration comments received: No.

Local Public Document Room location: Phoenix Public Library, 1221 N. Central Avenue, Phoenix, Arizona 85004

Duquesne Light Company, et al., Docket Nos. 50-334 and 50-412, Beaver Valley Power Station, Unit Nos. 1 and 2, Shippingport, Pennsylvania

Date of application for amendments: December 27, 1995

Brief description of amendments: These amendments modify Tables 3.3-11 and 4.3-7 of Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and BVPS-2) Technical Specification 3.3.3.8 (Accident Monitoring Instrumentation) such that only one valve position indication system for the power-operated relief valves and safety valves is required to be operable. Minor editorial changes to BVPS-1 TS 3.3.3.8 and its associated Action Statements are also being made. These changes make the requirements of TS 3.3.3.8 consistent with the NRC's Improved Standard Technical Specifications (NUREG-1431, Revision 1) and with the guidance of Regulatory Guide 1.97, NUREG-0578, and NUREG-0737.

Date of issuance: May 1, 1996

Effective date: As of the date of issuance, to be implemented within 60 days.

Amendment Nos.: 199 and 81

Facility Operating License Nos. DPR-66 and NPF-73: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: January 31, 1996 (61 FR 3499) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated May 1, 1996No significant hazards consideration comments received: No.

Local Public Document Room location: B. F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, Pennsylvania 15001.

Duquesne Light Company, et al., Docket No. 50-334, Beaver Valley Power Station, Unit No. 1, Shippingport, Pennsylvania

Date of application for amendment: February 12, 1996

Brief description of amendment: The amendment revises Technical Specification (TS) 4.6.2.2.d to delete the reference to the specific test acceptance criteria for the Containment Recirculation Spray Pumps and replaces the specific test acceptance criteria with reference to the requirements of the Inservice Testing (IST) Program. In addition, the 18-month test frequency is replaced with the test frequency requirements specified in the IST Program. The amendment also revises the Bases for TS 4.6.2.2.d to describe this revision to TS 4.6.2.2.d.

Date of issuance: May 7, 1996

Effective date: As of the date of issuance, to be implemented within 60 days.

Amendment No: 200

Facility Operating License No. DPR-66. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: March 13, 1996 (61 FR 10393) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 7, 1996 No significant hazards consideration comments received: No

Local Public Document Room location: B. F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, PA 15001

Florida Power Corporation, et al., Docket No. 50-302, Crystal River Nuclear Generating Plant, Unit No. 3, Citrus County, Florida

Date of application for amendments: March 21, 1996 as supplemented April 8, 15, and 18, 1996.

Description of amendment request: The proposed amendment provides for interim repair criteria for volumetric intergranular attack (IGA) indications in the once-through-steam generators (OTSG). The interim repair criteria is based on bobbin coil voltage response and motorized rotating pancake coil probe dimensional measurements. The amendment would be applicable for IGA indications within the region below the first tube support plate and the secondary face of the lower tubesheet (first span) of the OTSG and for one cycle only until Refuel 11.

Date of issuance: April 30, 1996

Effective date: April 30, 1996Amendment Nos. 154

Facility Operating License No. DPR-72: Amendment revised the Technical Specifications. Public comments requested as to proposed no significant hazards consideration: Yes (61 FR 13888). That notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by April 29, 1996, but indicated that if the Commission makes a final no significant hazards consideration determination any such hearing would take place after issuance of amendment. The Commission's related evaluation of this amendment is contained in a Safety Evaluation dated April 30, 1996

Local Public Document Room location: Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida 32629

Indiana Michigan Power Company, Docket No. 50-316, Donald C. Cook, Nuclear Plant, Unit No. 2, Berrien County, Michigan

Date of application for amendment: March 12, 1996 (AEP:NRC:1248)

Brief description of amendment: The amendment removes the technical specifications related to shutdown and control rod position indication while in shutdown modes 3, 4, and 5.

Date of issuance: May 2, 1996

Effective date: May 2, 1996, with full implementation within 45 days

Amendment No.: 194

Facility Operating License No. DPR-74. Amendment revises the Technical Specifications.

Date of initial notice in Federal Register: March 27, 1996 (61 FR 13527) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 2, 1996.No significant hazards consideration comments received: No.

Local Public Document Room location: Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, Michigan 49085

Nebraska Public Power District, Docket No. 50-298, Cooper Nuclear Station, Nemaha County, Nebraska

Date of amendment request: May 5, 1995 and July 14, 1995, supplemented by letter dated March 5, 1996

Brief description of amendment: The amendment revised the Technical Specifications to 1) verify that the redundant diesel generator is operable upon the loss of one diesel generator, and implement provisions to verify that the operable diesel generator does not have a common cause failure; 2) incorporate provisions to allow a modified start for the diesel generators; and 3) remove the requirement that the reactor power level be reduced to 25% of rated power upon loss of both diesel generator units or both incoming power sources (start-up and emergency transformers). In addition, the period of time allowed for continued reactor operation with both diesels inoperable was reduced from 24 to two hours.

Date of issuance: April 29, 1996

Effective date: April 29, 1996

Amendment No.: 175

Facility Operating License No. DPR-46: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: September 27, 1995 (60 FR 49939) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 29, 1996.No significant hazards consideration comments received: No.

Local Public Document Room location: Auburn Memorial Library, 1810 Courthouse Avenue, Auburn, NE 68305.

North Atlantic Energy Service Corporation, Docket No. 50-443, Seabrook Station, Unit No. 1, Rockingham County, New Hampshire

Date of amendment request: September 22, 1995

Description of amendment request: The amendment changes the ACTION specified in Table 3.3-3, Engineered Safety Features Actuation System Instrumentation, from ACTION 18 to ACTION 15 for Functional Unit 8.b, Automatic Switchover to Containment Sump - RWST Level Low-Low.

Date of issuance: May 7, 1996,

Effective date: As of the date of issuance, to be implemented within 60 days.

Amendment No.: 47

Facility Operating License No. NPF-86. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: December 6, 1995 (60 FR 62493) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 7, 1996. No significant hazards consideration comments received: No.

Local Public Document Room location: Exeter Public Library, Founders Park, Exeter, NH 03833.

Northeast Nuclear Energy Company, et al., Docket No. 50-336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of application for amendment: May 26, 1995, as supplemented October 20, 1995, and May 3, 1996.

Brief description of amendment: The amendment modifies Technical Specification (TS) 3.8.1.2, ''Electrical Power Systems, Shutdown,'' TS 3.8.2.2, ''Electrical Power Systems, A.C. Distribution - Shutdown,'' and TS 3.8.2.4, ''Electrical Power Systems, D.C. Distribution - Shutdown,'' to provide operational flexibility as well as consistency between action statements and to eliminate certain surveillance requirements that are not applicable in Mode 5 or 6.

The proposed changes relating to TS 3.8.1.1, ''Electrical Power Systems, A.C. Sources, Operating,'' are not included in this amendment since this portion of the TS change is still under review by the staff and will be addressed at a later date.

Date of issuance: May 6, 1996

Effective date: As of the date of issuance, to be implemented within 30 days.

Amendment No.: 197

Facility Operating License No. DPR-65. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: December 6, 1995 (60 FR 62493) The October 20, 1995, letter formally withdrew the need for exigent handling of the May 26, 1995, request and requested an additional change to TS 3.8.2.4. The May 3, 1996, letter withdrew a portion of the initial request which did not affect the initial proposed no significant hazards consideration. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 6, 1996. No significant hazards consideration comments received: No.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, CT 06360, and Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, CT 06385.

Power Authority of the State of New York, Docket No. 50-286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of application for amendment: March 14, 1996

Brief description of amendment: The amendment allows a one-time extension of the intervals for the pressurizer safety valve setpoint and snubber functional testing that is due in May 1996.

Date of issuance: May 3, 1996

Effective date: As of the date of issuance to be implemented within 30 days.

Amendment No.: 165

Facility Operating License No. DPR-26: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: April 3, 1996, (61 FR 14835) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 3, 1996.No significant hazards consideration comments received: No

Local Public Document Room location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10610.

Public Service Electric & Gas Company, Docket Nos. 50-272 and 50-311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of application for amendments: January 4, 1996

Brief description of amendments: The amendments change Technical Specification 3/4.8.2.5, ''28-Volt D.C. Distribution - Operating.'' The amendment for Unit 1 makes Unit 1 requirements similar to Unit 2 by defining the specific battery chargers that are required for each train and by restricting the use of the backup battery charger to 7 days. The amendments for both units also require that the 28-Volt DC bus be energized for that bus to be OPERABLE.

Date of issuance: April 29, 1996

Effective date: Both units, as of date of issuance, to be implemented within 60 days.Amendment Nos. 182 and 163

Facility Operating License Nos. DPR-70 and DPR-75. The amendments revised the Technical Specifications.

Date of initial notice in Federal Register: February 14, 1996 (61 FR 5818) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 29, 1996No significant hazards consideration comments received: No

Local Public Document Room location: Salem Free Public Library, 112 West Broadway, Salem, New Jersey 08079

Union Electric Company, Docket No. 50-483, Callaway Plant, Unit 1, Callaway County, Missouri

Date of application for amendment: September 6, 1995, as supplemented by letters dated January 30, 1996, March 27, 1996, and April 2, 1996.

Brief description of amendment: The amendment revises TS 5.3.1 to reflect a change in the maximum initial enrichment for reload fuel, subject to the integral fuel burnable absorber (IFBA) requirements, and a change in the maximum fuel enrichment not requiring IFBAs. The amendment also changes the maximum reference $k_{infinity}$ in TS 5.6.1.1 for fuel storage in Region 1 of the spent fuel pool and revises TS Figure 3.9-1 to reflect a change to the maximum initial enrichment for fuel stored in Region 2 of the spent fuel pool.

Date of issuance: April 30, 1996

Effective date: April 30, 1996, to be implemented within 30 days from the date of issuance.

Amendment No.: 109

Facility Operating License No. NPF-30: The amendment revised the Technical Specifications.

Date of initial notice in Federal Register: November 8, 1995 (60 FR 56372). The January 30, 1996, March 27, 1996, and April 2, 1996, supplemental letters provided additional clarifying information and did not change the original no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 30, 1996.No significant hazards consideration comments received: No.

Local Public Document Room location: Callaway County Public Library, 710 Court Street, Fulton, Missouri 65251.

Union Electric Company, Docket No. 50-483, Callaway Plant, Unit 1, Callaway County, Missouri

Date of application for amendment: February 9, 1996

Brief description of amendment: The amendment revised Technical Specification 5.3.1 to allow the use of ZIRLO clad fuel rods and ZIRLO filler rods.

Date of issuance: April 30, 1996

Effective date: April 30, 1996, to be implemented within 30 days of issuance.

Amendment No.: 110

Facility Operating License No. NPF-30: The amendment revised the Technical Specifications.

Date of initial notice in Federal Register: February 28, 1996 (61 FR 7558) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 30, 1996.No significant hazards consideration comments received: No.

Local Public Document Room location: Callaway County Public Library, 710 Court Street, Fulton, Missouri 65251.

Virginia Electric and Power Company, Docket Nos. 50-280 and 50-281, Surry Power Station, Unit Nos. 1 and 2, Surry County, Virginia.

Date of application for amendments: January 30, 1996

Brief description of amendments: These amendments modify the Technical Specifications requirements for the sampling of the reactor coolant for dissolved oxygen chlorides and fluorides.

Date of issuance: 209 and 209

Effective date: April 29, 1996

Amendment Nos. 209 and 209

Facility Operating License Nos. DPR-32 and DPR-37: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: March 27, 1996 (61 FR 13533) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 29, 1996.No significant hazards consideration comments received: No

Local Public Document Room location: Swem Library, College of William and Mary, Williamsburg, Virginia 23185

Washington Public Power Supply System, Docket No. 50-397, Nuclear Project No. 2, Benton County, Washington

Date of application for amendment: January 19, 1996, as supplemented by letter dated March 19, 1996.

Brief description of amendment: The amendment modifies the Technical Specifications for leak tests of containment isolation valves. The amendment replaces the current specified surveillance intervals for containment leak testing with new surveillance requirements to conduct containment leak testing according to a performance-based containment leak test program.

Date of issuance: May 8, 1996

Effective date: May 8, 1996, to be implemented within 30 days of issuance.

Amendment No.: 144

Facility Operating License No. NPF-21: The amendment revised the Technical Specifications.

Date of initial notice in Federal Register: February 14, 1996 (61 FR 5820) The March 19, 1996, supplemental letter provided additional clarifying information and did not change the initial no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated May 8, 1996.No significant hazards consideration comments received: No.

Local Public Document Room location: Richland Public Library, 955 Northgate Street, Richland, Washington 99352

Dated at Rockville, Maryland, this 15th day of May 1996.

For the Nuclear Regulatory Commission

Steven A. Varga,

Director, Division of Reactor Projects - I/II, Office of Nuclear Reactor Regulation

[Doc. 96-12691 Filed 5-21-96; 8:45 am]

BILLING CODE 7590-01-F

OFFICE OF THE UNITED STATES TRADE REPRESENTATIVE

Notice of Industry Policy and Sector/Functional Advisory Committee Meetings

AGENCY: Office of the United States Trade Representative.

ACTION: Notice of Industry Policy and Sector/Functional Advisory Committee meetings.

SUMMARY: The meetings will include a review and discussion of current issues which influence U.S. trade policy. Pursuant to section 2155(f)(2) of title 19 of the United States Code, the U.S. Trade Representative has determined that these meetings will be concerned with matters the disclosure of which would seriously compromise the Government's negotiating objectives or bargaining positions. Accordingly, these meetings will be closed to the public.

DATES: The period of March 1, 1996 to March 1, 1998.

ADDRESSES: All meetings will be held at the U.S. Department of Commerce, 14th Street and Independence Avenue, Washington, D.C. 20230, unless an alternate site is necessary.

FOR FURTHER INFORMATION CONTACT: Ms. Phyllis Shearer Jones, Assistant U.S. Trade Representative for Intergovernmental Affairs and Public Liaison, Office of the United States Trade Representative at (202) 395–6120 or Wendy Smith, Director of the Trade Advisory Center, Department of Commerce at (202) 482–3268.

Charlene Barshefsky,

Acting United States Trade Representative.

[FR Doc. 96–12858 Filed 5–21–96; 8:45 am]

BILLING CODE 3190–01–M

PRESIDENTIAL ADVISORY COMMITTEE ON GULF WAR VETERANS' ILLNESSES

Notice of Open Meeting

AGENCY: Presidential Advisory Committee on Gulf War Veterans' Illnesses.

SUMMARY: This notice is hereby given to announce an open meeting of a panel of the Presidential Advisory Committee on Gulf War Veterans' Illnesses. The panel will discuss scientific and clinical issues related to reproductive health and Gulf War veterans and will receive comment from members of the public. Dr. Joyce C. Lashof, Advisory Committee chair, will chair this panel meeting.

DATES: June 17, 1996, 9:30 a.m.–4:15 p.m.; June 18, 1996, 8:30 a.m.–12:30 p.m.

PLACE: Renaissance Madison Hotel, 515 Madison Street, Seattle, WA 98104.

SUPPLEMENTARY INFORMATION: The President established the Presidential Advisory Committee on Gulf War Veterans' Illnesses by Executive Order 12961, May 26, 1995. The purpose of this Advisory Committee is to review and provide recommendations on the full range of government activities associated with Gulf War veterans' illnesses. The Advisory Committee reports to the President through the Secretary of Defense, the Secretary of Health and Human Services, and the Secretary of Veterans Affairs. Advisory Committee members have expertise relevant to the functions of the Committee and are appointed by the President from non-Federal sectors.

Tentative Agenda

Monday, June 17, 1996.

9:30 a.m. Call to order and opening remarks
9:40 a.m. Public comment
10:40 a.m. Break
11:00 a.m. Public comment (cont.)
12:00 .m. Lunch
1:15 p.m. Biological plausibility: teratology, ovarian toxicity, and spermatotoxicity
2:00 p.m. Reproductive toxicology, hazard assessment, and the Gulf War
2:45 p.m. Break
3:00 p.m. Epidemiology of infertility, subfertility, fetal loss, and birth defects in the U.S.
3:35 p.m. Evaluating rates of congenital anomalies in children of Gulf War veterans
4:15 p.m. Recess

Tuesday, June 18, 1996

8:30 a.m. Call to order